APPENDIX VIII: INTEGRATED GEN IV MATERIALS TECHNOLOGY DEVELOPMENT

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VIII.1 INTEGRATED MATERIALS TECHNOLOGY DEVELOPMENT

An integrated R&D program will be conducted to study, quantify, and in some cases, develop materials with required properties for the Gen IV advanced reactor systems. The goal of the National Materials Crosscut Program (NMCP) is to ensure that the required Gen IV materials R&D will be a comprehensive and integrated effort to identify and provide the materials data and its interpretation needed for the design and construction of the selected advanced reactor concepts.

For the range of service conditions expected in Gen IV systems, including possible accident scenarios, sufficient data must be developed to demonstrate that the candidate materials meet the following design objectives:

- acceptable dimensional stability including void swelling, thermal creep, irradiation creep, stress relaxation, and growth;
- acceptable strength, ductility, and toughness;
- acceptable resistance to creep rupture, fatigue cracking, creep-fatigue interactions, and helium embrittlement; and
- acceptable chemical compatibility and corrosion resistance (including stress corrosion cracking and irradiation-assisted stress corrosion cracking) in the presence of coolants and process fluids.

Additionally, it will be necessary to develop validated models of microstructure-property relationships to enable predictions of long-term materials behavior to be made with confidence and to develop high-temperature materials design methodology for materials, use, codification, and regulatory acceptance.

To make efficient use of program resources, the development of the required databases and methods for their application must incorporate both the extensive results from historic and ongoing programs in the United States and abroad that address related materials needs. These would include, but not be limited to, DOE, NRC, and industry programs on liquid-metal-, gas-, and light-water-cooled reactor, fossil-energy, and fusion materials research programs, as well as similar foreign efforts.

Since many of the challenges and potential solutions will be shared by more than one reactor concept, it will be necessary to work with the system integration managers (SIMs) for each individual reactor concept to examine the range of requirements for its major components to ascertain what the materials challenges and solutions to those will be and then establish an appropriate breakdown of responsibilities for the widely varying materials needs within the Gen IV Initiative. It is expected that there will be two primary categories for materials research needs:

- Materials needs that crosscut two or more specific reactor system and
- Materials needs specific to one particular reactor concept or energy conversion technology.

Where there are commonly identified materials needs for more than one system, it will be appropriate to establish a crosscutting technology development activity to address those issues. Where a specific reactor concept has unique materials challenges, it will be appropriate to address those activities in conjunction with that particular reactor systems's R&D. Examples of this category of materials needs include reactor-specific materials compatibility issues associated with a particular coolant and materials used within only one reactor concept (i.e., graphite within the VHTR).

The National Materials Program within the Gen IV Initiative will have responsibility for establishing and executing an integrated plan that addresses cross-cutting, reactor-specific, and energy-conversion materials research needs in a coordinated and prioritized manner.

Four interrelated areas of materials R&D are generally considered crosscutting: (1) qualification of materials for service within the vessel and core of the reactors that must withstand radiation-induced challenges; (2) qualification of materials for service in the balance of plant that must withstand high-temperature challenges; (3) the development of validated models for predicting long-term, physically based microstructure-property relationships for the high-temperatures, extended-operation periods, and high irradiation doses that will exist in Gen IV reactors; and (4) the development of an adequate high-temperature-materials design methodology to provide a basis for design, use, and codification of materials under combined time-independent and time-dependent loadings.

Reactor-specific materials research that has been identified for the individual reactor and energy-conversion concepts includes materials compatibility with a particular coolant or heat-transfer medium, as well as materials expected to used only within a single reactor or energy conversion system, such as graphite, selectively permeable membranes, catalysts, etc. A special category of reactor-specific materials research will also include research that must be performed at pace that would significantly precede normal cross-cutting research in the same area (e.g. NGNP reactor system materials R&D).

While the current plan addresses materials issues for all the reactors currently being examined within the Gen IV program, there is recognition that the plans to build a VHTR as the Next Generation Nuclear Plant (NGNP) by 2017 will strongly drive much of the materials research during the next ten years of the program. Accordingly, though the four crosscutting activities described below will include materials of interest to all the reactors, where possible, the emphasis will be on materials that meet the needs of the NGNP, while at the same time supporting the other reactor concepts. Where the NGNP materials needs clearly outstrip those of the other reactor systems, they will be addressed independently and the other reactor systems will be able to utilize those results that are relevant.

A final category of materials R&D that is recognized within the Gen IV Program is that which overlaps the materials needs for the development of fuels and reprocessing technology within the Advanced Fuel Cycle Initiative (AFCI) and for chemical processing equipment for the Nuclear Hydrogen Initiative (NHI). While both AFCI and NHI are independent programs with their own research objectives and funding, it has already been recognized their applications will contain many of the same conditions that exist for reactor systems and their components in the Gen IV Program and, hence, may utilize a common set of structural materials. A special involvement among all three programs is being developed and maintained to help ensure that the materials R&D being conducted within them is coordinated to minimize duplication and costs and maximize mutually beneficial materials technology development and qualification.

VIII.2 MATERIALS CROSSCUTTING TASKS

VIII.2.1 Materials for Radiation Service

The performance of structural materials is limited, in general, by the degradation of physical and mechanical properties by exposure to energetic neutrons or by exposure to the chemical environment provided by the primary coolant medium. Although there are very significant differences in operating environments between the various concepts under consideration, it is possible to identify a number of common environmental features. Of these common features, operating temperatures and neutron exposures will have the greatest impact on materials performance and component lifetimes, leaving aside for the moment the issues surrounding radiation-assisted corrosion phenomena. Therefore, combining the evaluation of materials as a function of neutron exposure offers an opportunity for addressing the development and qualification of materials for multiple concepts within a coordinated set of irradiation experiments. Evaluation of candidate materials that are applicable for multiple concepts offers both an improved overall database and the potential for significant cost savings compared to conducting separate irradiation programs for each reactor concept. A prime example would be the design and implementation of an irradiation program that would simultaneously serve the needs for an irradiation effects database for many of the Gen IV reactors.

A second important crosscutting feature to be considered is that data on radiation effects must be obtained for all Gen IV reactor concepts from a limited set of operating test reactors and irradiation facilities; for example, HFIR and ATR in the US, HFR in the EU, BOR60 in Russia, and JMTR and JOYO in Japan. Significant opportunities exist for the sharing of information on the technology of irradiation testing, specimen miniaturization, advanced methods of property measurement, and the development of a common materials property database system that would crosscut all potential reactor concepts. Although it is possible in a limited number of cases to provide an irradiation test environment that is prototypical for some of the components of a particular Gen IV concept, irradiation test conditions are generally non-prototypical, either because the required spectral conditions cannot be achieved or the required neutron lifetime exposures can only be achieved by testing at accelerated dose rates. Additionally, individual components experience spatial variations in flux, spectrum and mechanical loading. Of necessity, materials selection will have to be based upon incomplete experimental databases and consequently there is a strong and cross-cutting need for the development of physically-based models of critical radiation effects phenomena in both FCC and BCC alloy systems based upon advanced microstructural analysis. Such validated models are needed to provide a sound basis for making extrapolations and interpolations from the experimental radiation effects database. While the development of such models will be conducted within a separate crosscutting task focused on that area, the development of the experimental databases upon which those models will be based will be responsibility of work within this task.

A final important thread that links the structural materials various Gen IV in-vessel components is that several classes of structural alloys find application in more than one system. Examples include creep-resistant low-swelling austenitic stainless steels and ferritic/martensitic steels for in-vessel components for the SCWR, GFR and LFR and nickel-based alloys for the NGNP and MSR internals. For very high temperature applications, refractory metal alloys and structural composites such as SiC/SiC could have potential applications in the long term for more than one concept. Within the rapidly evolving field of mechanically alloyed materials, oxide-dispersion-strengthened (ODS) alloys based on austenitic, ferritic or ferritic/martensitic matrices have the potential to significantly advance the performance of components for all the primary Gen IV concepts under consideration. Programs to develop ODS materials for nuclear applications are being strongly pursued in Europe and in Japan. Efforts to understand the processing-microstructure-property relationships for mechanically alloyed materials could eventually lead to the development of alloys with exceptional high-temperature creep strength, microstructural stability, outstanding resistance to void swelling, and the ability to retain properties following off-normal temperature excursions.

The activities and funding within this crosscutting task and its associated milestones included in section VIII.8 are expected to address general needs of materials for radiation service for the GFR, LFR, and SCWR systems. Specialized schedule-driven reactor-specific needs for NGNP system materials for radiation service are addressed in section VIII.3.1 on NGNP materials. Specialized reactor-specific needs for SCWR system materials for radiation service that must additionally address stress-corrosion corrosion cracking and irradiation-assisted stress-corrosion cracking are addressed in section VIII.3.3 on SCWR materials. Funding for the reactor-specific NGNP and SCWR irradiated materials research is included within the materials funding requirements in Appendices I and III, respectively.

VIII.2.2 Materials for High-Temperature Service

In the Gen IV Initiative, although the operating conditions vary significantly from one reactor system to the next, significant commonality exists with regard to the selection of materials for their high-temperature structural components. These common issues can advantageously be addressed in a crosscutting task. However, in setting out the scope and schedule of this crosscutting task, it is recognized that the highest priority for development and qualification of materials for high-temperature service is given to NGNP, as the first candidate system to be deployed. It follows that in qualification of materials for high-temperature service, early crosscutting efforts will be focused mainly on establishing the activities that will complement those being pursued for NGNP to establish a sound foundation for the multiple Gen IV reactor systems. This will pave the way for the crosscutting activities to gradually increase in scope as portions of the NGNP efforts approach completion.

The crosscutting materials evaluation and qualification activities will be initiated at the early stage of the program and gradually phase in to follow the development of the leading task on NGNP materials qualification. Analysis indicates that despite of the various operating conditions in the proposed reactor systems, significant commonality exists with regard to the selection of materials for their high-temperature structural components. As a result, the materials for Class I nuclear components for service above the temperature limits of ASME Section III will be limited to those materials incorporated into Section III, Subsection NH. Currently, this subsection permits construction with a very few alloys, namely type 304H and type 316H stainless steels, alloy 800H, and 2 1/4Cr-1Mo steel (class 1). The incorporation of Gr91 (modified 9Cr-1Mo-V) steel is in progress. To take full advantage of the potential of the reactor concepts in the Gen IV Initiative, it will be necessary to utilize the advances made in the structural materials technology, select the most promising candidate materials for higher temperature service, and move forward toward acceptance of these materials into the appropriate construction codes.

Even though many of the materials that will be required for construction of high-temperature, out-of-core components will be the same as those used for some in-core applications, the focus of this crosscutting technology development task will be on their unirradiated high-temperature qualification. While short-term tensile and fatigue properties will need to be evaluated for these materials, it is time-dependent creep and creep-fatigue, which are the primary limitations for materials use, that will be most strongly limiting and, hence, principally addressed. The crosscutting technology development associated with high-temperature use of these materials in the presence of neutron irradiation will be addressed in the task on Qualification of Materials for Radiation Service described in Section 5.9.1. The related high-temperature corrosion and compatibility concerns for these materials will be addressed as part of reactor-specific R&D tasks and discussed, in summary, in section 5.9.5.1.

For the high-temperature materials to be evaluated for out-of-core applications for the Gen IV initiative, the destination of this crosscutting materials research thrust will be their eventual incorporation into ASME Section III, Subsection NH. The materials for such high-temperature service may be separated into several categories by approximate upper-use temperatures. While there is some overlap, and more advanced materials within a class will somewhat extend the temperature limits of current materials, these classes roughly correspond to: (a) ferritic steels including bainitic and martensitic steels up to 12% chromium for use up to about 650°C, (b) austenitic stainless steels for use up to about 800°C, (c) high alloys, in which iron content is greater than any other element, and nickel-base alloys for use up to about 900-950°C, and (d) special materials such as oxide dispersion-strengthened (ODS) alloys for possible use up to about 1000-1050°C.

The two primary thrusts within this crosscutting activity in the first ten years of the Gen IV Initiative will be to: (1) evaluate the current commercial or near-commercial materials for adequacy of data and properties to incorporate into Subsection NH of the ASME Section III for high-temperature service and begin the codification of those appropriate materials, including generation of incremental required data bases, and (2) perform evaluation and screening of promising advanced materials for higher temperature service, resulting in the selection of candidate materials for further development and eventual inclusion into the Section III Subsection NH. These evaluation and development activities will include all appropriate product forms and section thicknesses needed for required reactor components, including weldments and their constituents (weldmetal, HAZ, and basemetal). Since the crosscutting activity involves Gen IV reactor systems with later anticipated deployment dates than that of the NGNP, more efforts for evaluation of advanced materials for high-temperature service can be included.

The activities and funding within this crosscutting task and its associated milestones included in section VIII.8 are expected to address general needs of materials for high-temperature service for the GFR, LFR, and SCWR systems. Specialized schedule-driven reactor-specific needs for NGNP system materials for high-temperature service are addressed in section VIII.3.1 on NGNP materials. Specialized reactor-specific needs may be identified for the GFR system for non-metallic materials for high-temperature service that are beyond the scope of this crosscutting task, but the GFR materials requirements are still being developed. Funding for the reactor-specific NGNP and GFR high-temperature materials research will be included within the materials funding requirements in Appendices I and II, respectively.

VIII.2.3 Development of Microstructure-Properties Models

For each design objective described in section VIII.1, the development and evolution of the fundamental microstructural features that establish materials performance need to be understood to further improve material performance and/or ensure the very long operational life envisioned for Gen IV reactor systems. This understanding will require a combination of theory and modeling activities tied to detailed microstructural characterization and mechanical property measurements. The models must be developed using the best current materials science practice in order to provide a sound basis for interpolating and extrapolating materials performance beyond experimental data bases, as well as providing the fundamental understanding needed to make designed changes in material compositions and processing to achieve improved properties.

At the recent Higher Temperature Reactor Materials Workshop, Sponsored by the DOE Offices of Nuclear Energy, Science, and Technology and Basic Energy Sciences in March of 2002, the issues associated with microstructural development and modeling were extensively discussed. Significant conclusions from the meeting, including needs for the Gen IV Reactor Initiative, are:

- Displacement damage during irradiation creates a non-equilibrium, structure-chemistry evolution at the nanoscale that alters plasticity, corrosion-oxidation and fracture processes. The crucial elements of the microstructure that evolve with irradiation are voids and bubbles, dislocation loops and stacking fault tetrahedra, carbides and other precipitates, and network dislocations. Radiation-induced solute segregation (RIS) can lead to the formation of unexpected phases in the matrix, and composition changes at free surfaces and interior interfaces. RIS influences both mechanical properties and corrosion behavior. In addition, the diffusion and segregation of helium and hydrogen to vacancy clusters and voids is a major contributor to swelling. Fundamental understanding of these complex, interdependent, radiation-induced material changes is essential to underpin the development of Gen IV reactor systems.

- The key structural performance issues for most irradiated metallic alloys are hardening-induced embrittlement at low temperatures, and time-dependent deformation (creep and fatigue) and cracking at high temperatures. The evolution of non-equilibrium structures and chemistries promote a hardened matrix and lower grain-boundary cohesive strengths, thereby reducing the tensile stress required for cleavage or intergranular fracture. At high temperatures, the radiation-induced changes in the matrix and particularly at grain boundaries can promote creep embrittlement. The atomistics of fracture need to be combined with micromechanical models to better elucidate behavior in complex, radiation-induced, multi-component nanostructures.

A series of integrated, physically based, empirically validated models will need to be developed to address the issues raised above, guide overall materials development, and ensure long-term materials stability during operation. Six general topics will need to be addressed.

- Development of improved fundamental understanding and modeling of the nucleation-phase of the various defect types that are produced during irradiation (e.g. vacancy and interstitial aggregates, second phases, etc.);
- Development of atomistic and continuum models that describe the mechanisms responsible for radiation-enhanced, -induced, and -modified microstructural changes and the physical phenomena that account for the persistence of those microstructures that remain stable at high temperatures;
- Development of the kinetic and thermodynamic models required to provide an understanding of the formation and stability, particularly under irradiation, of both undesirable and desirable second phase precipitates. A critical example of a desirable second phase is the very fine oxide clusters that provide the high-temperature strength of ODS alloys;
- Development of improved micromechanical models to investigate the detailed interactions between dislocations and other microstructural features which control material strength and deformation behavior. Detailed atomistic modeling is required to provide parameters and insight for higher level deformation models;
- Development of improved understanding for the mechanisms that contribute to high-temperature, time-dependent plasticity (e.g. creep-fatigue, racheting, etc.) and the models describing them for application and insight into the Improved High-Temperature Design Methodology to be developed under a separate crosscutting task; and

- Performance of detailed microstructural analysis, down to the atomic scale, on Gen IV candidate materials using state-of-the-art characterization techniques (e.g. atom probe, X-ray and small-angle neutron scattering, positron annihilation, high-resolution transmission electron microscopy, etc.) to provide microstructural input for model development.

Although the detailed microstructural analysis required for model development may be carried out as part of this task, it is anticipated that the samples for examination will be obtained from materials irradiated in experiments carried out under other tasks, particularly those on Qualification of Materials for Radiation Service and Reactor-Specific Materials. In some cases, special-purpose experiments may be proposed and conducted as part of this effort.

The activities and funding within this crosscutting task and its associated milestones included in section VIII.8 are expected to address the anticipated microstructural analysis and model development needs for all Gen IV reactor systems.

VIII.2.4 Development of Improved High-Temperature Design Methodology

The objective of the High-Temperature Design Methodology Task is to establish the improved and expanded structural design technology necessary to support the codification and utilization of structural materials in high-temperature Gen IV reactor system components. The temperatures and materials requirements of most Gen IV components exceed the time/temperature coverage currently provided by Subsection NH of Section III of the ASME Boiler and Pressure Vessel Code, which governs the design and construction of elevated-temperature, Class 1 nuclear components. This task will provide the data and models required by ASME Code groups to formulate time-dependent failure criteria and assessment rules and procedures that will ensure adequate life for components fabricated from the metallic alloys chosen for Gen IV systems. The task will also provide the material behavior (constitutive) models for the detailed inelastic design analysis methods required by Subsection NH for accessing critical structural regions, and, it will provide the simplified inelastic design analysis methods that are allowed for less critical regions and are used for preliminary design.

Subsection NH of the ASME Code currently covers just four high-temperature alloys: 304 and 316 stainless steel, 2-1/4 Cr-1Mo steel, and Alloy 800H. Modified 9Cr-1Mo steel (Grade 91) has been approved but is not yet included. The maximum temperature coverage for these materials is inadequate for NGNP, GFR, and LFR (long-term version). In addition, the maximum design life allowed is, at most, 34 years whereas Gen IV components are to have a design life of 60 years. Thus, most Gen IV systems will require the inclusion in Subsection NH of new materials with higher permitted temperatures and longer operating times. Even for systems and components operating within the range of coverage of Subsection NH, new stronger materials may be desirable, and, in any event, the time coverage must be significantly increased.

Candidate structural materials for Gen IV systems fall primarily into two classes: medium high-temperature alloys, characterized by the Cr-Mo steels and AISI 304 and 316 stainless steel, and very high-temperature alloys, characterized by nickel-base alloys. The strategy for the crosscutting development effort is to focus initial efforts on a single representative and promising material from each class—modified 9Cr-1Mo steel (Grade 92) at medium high temperatures and nickel-base Alloy 617 at very high temperatures. As other key structural alloys are identified for the various reactor concepts, they will be factored into the effort, especially where an identified material is common to more than one reactor concept. The developments for modified 9Cr-1Mo steel (Grade 92) and Alloy 617 will provide an initial focus for Code work, and the resulting criteria, design analysis, and assessment methods will provide the framework and springboard for introducing additional materials as they are identified. They will also provide the near-term tools needed by NGNP designers to develop conceptual and preliminary designs.

A unique requirement for most Gen IV materials is that they will operate at the upper end of their useful temperature range. At the lower end of a material's useful elevated-temperature operating range, the inelastic response to cyclic thermal and mechanical loadings, especially at discontinuities, can usually be separated into time-dependent plasticity and time-dependent creep. Current Subsection NH rules and criteria, as well as the associated inelastic design analysis methods and simplified methods, depend heavily on this assumed separation. At higher temperatures, the distinction between rate-independent plasticity and time-dependent creep blurs for many materials (e.g., modified 9Cr-1Mo steel, Grade 91, and Alloy 617), and the separation between behaviors is no longer valid. The response becomes very rate dependent, and both strain-and cyclic-softening occur. The criteria and analysis methods for Gen IV components must be formulated to reflect these behavioral features.

The High-Temperature Design Methodology Task has several subtasks. The first is the development of experimentally-based constitutive equations required for inelastic design analyses. These equations, which will be developed for each key material, starting with modified 9Cr-1Mo steel (Grade 92) and Alloy 617, will be unified, in the sense that they will not distinguish between rate-dependent plasticity and time-dependent creep.

The second subtask, which will be carried out in close coordination with the ASME Code Subgroup on Elevated Temperature Design, is the development of failure models for design criteria. These models, which again will be experimentally based, normally consist of two parts: (1) a damage accumulation model describing failure resulting from the accumulation of damage under time-varying thermal and mechanical loadings, and (2) a strength criterion describing failures under multiaxial stresses. Major challenges of this subtask are developing an adequate treatment for creep-fatigue failures, especially at very high temperatures, and an improved means of addressing notches and weldments (both major unresolved NRC concerns in the Clinch River Breeder Reactor Plant licensing process).

Perhaps the most challenging subtask will be the development of simplified methods. While the underlying premise of Subsection NH is that the variation of stresses and strains with time in a high-temperature component should be predicted by detailed inelastic design analyses, the wide use of such analyses for preliminary design and for every region and loading condition of a component would prove impracticable. Thus, limited simplified rules for satisfying strain limits (ratcheting) and creep-fatigue criteria are included in Subsection NH. However, at the upper end of a material's operating range the material response previously described violates basic assumptions used in developing the existing simplified methods. Thus, new methods must be developed, and quickly, since they are required in the early stages of design.

The final two subtasks are (1) confirmatory structural tests and (2) procedures for safety and reliability assessments. The role of structural tests, which will involve the determination of deformation and failure behavior in generic features as opposed to actual components) is to either validate the high-temperature structural design methodology or, if that does not occur, to guide required improvements. The safety/reliability subtask will focus on the safety assessment methodology that will be required for licensing. Included will be a high-temperature flaw growth and assessment procedure and a criterion for ultimate structural failure.

The activities and funding within this crosscutting task and its associated milestones included in section VIII.8 are expected to address the high-temperature design methodology needs for materials for the GFR, LFR, and SCWR systems. Specialized schedule-driven reactor-specific needs the development of high-temperature design methodology for NGNP system materials is addressed in section VIII.3.1 on NGNP materials. Funding for the reactor-specific NGNP materials research is included within the materials funding requirements in Appendix I.

VIII.3 REACTOR-SPECIFIC MATERIALS

Reactor-specific materials research includes materials compatibility with a particular coolant or heat-transfer medium used in a single reactor system, as well as structural materials expected to used only within a single reactor or energy conversion system, such as graphite, selectively permeable membranes, catalysts, etc. Additionally, where research must be performed at pace that would significantly precede cross-cutting research in the same area (e.g. NGNP reactor system materials R&D), it has also been classified as being reactor-specific.

Reactor-specific research identified to date is described for each reactor system in the sections that follow. Materials needs for the NGNP and SCWR have been fairly well addressed in the past year and those needs that are not addressed in the crosscutting tasks described above are summarized below. Detailed materials needs assessments for the GFR and LFR systems have not yet been performed, but in the areas where those needs have been identified, they are included herein. Future revisions of this appendix are expected to expand upon the materials needs for those systems.

While limited funding has been provided for a small crosscutting task established to provide coordination of reactor-specific materials research, the funding for the actual research, development, and qualification of reactor specific materials is included within the materials funding requirements within Appendices I to IV.

VIII.3.1 NGNP Reactor-Specific Materials

VIII.3.1.1 NGNP Materials for Radiation Service

The reactor pressure vessel (RPV) system will comprise a large RPV containing the core and internals, a second large power conversion vessel (PCV) containing the main turbine, generator, and associated turbo machinery and heat exchangers, and a pressure-containing cross vessel (CV) joining the RPV and the PCV. Because of the wide range of material thicknesses in the RPV, it will be constructed in a segmented configuration. Although the specific design is not yet available, such a configuration will play a role in the materials selection as it relates to fabrication issues, effects of loading variables such as cycling, etc. The vessels will be exposed to air on the outside and helium on the inside, with emissivity of the chosen material an important factor regarding radiation of heat from the component to the surroundings to ensure adequate cooling during accident conditions. The materials tentatively selected for gas-cooled RPV service are low-alloy ferritic/martensitic steels, alloyed primarily with chromium and molybdenum. The most significant difference in demands placed on the RPV system between previous gas-cooled reactor designs and the NGNP are the temperatures at which they will be required to operate, with the currently anticipated temperatures for the NGNP RPV and CV being about 650ºC under normal conditions and up to 770ºC under abnormal conditions (for times up to ~50 h). Because of the high operating temperature, low Cr-Mo steels such as 2 1/4Cr-1Mo do not have adequate high-temperature strength for the RPV and CV. In fact, the current operating temperature of 650ºC for the RPV and CV is at the limit for any ferritic or ferritic-martensitic steel currently in any part of the ASME B&PV Code, while the abnormal (off-normal accident) temperature of 770ºC for 50 hours is beyond that limit and provides an even greater challenge. Regarding irradiation exposure, current estimates for the RPV is about 3×10^{19} n/cm^2 (>0.1 MeV) for 60 y, about 0.0075 displacements per atom (dpa), approximately one-third that for typical current generation light-water reactors (LWR).

Potential candidate alloys for the PCV could include those for the RPV and CV, but there are lower cost options available because of the lower operating temperatures. Even under abnormal conditions, the PCV will be subjected to temperatures about the same as those currently used for commercial LWR vessels (~300ºC). Moreover, the size of the vessel is well within normal fabrication capability. Thus, the current LWR pressure vessel materials, SA508 grade 3 class 1 forgings or SA533 grade B class 1 plates are potential candidates, as is the 2 1/4Cr-1 Mo alloy, dependent on material compatibility issues. Also, the estimated irradiation exposure for closure bolting will be assessed to evaluate the need for inclusion of bolting in the irradiation program.

In the case of reactor metallic internals, depending on the specific component, the normal operating temperatures will range from 600 to 1000°C. However, the maximum temperature estimated for accident conditions ranges from 600 to 1200°C from one component to another. Additionally, radiation and thermal aging effects on properties are important considerations in material selection. Currently anticipated irradiation exposure levels for these components are quite low and radiation exposure, in general, is not expected to present a major challenge to the potential materials. Potential candidate materials for the internals, as well as the other high-temperature components likely to be constructed from metallic alloys, include a wide range of high alloy steels and nickel base alloys. These materials include alloys for which significant databases exist and new state-of-the-art alloys which are being developed for other high-temperature applications. For very-high-temperature components (>760°C), the most likely material candidates are:

1. Variants or restricted chemistry versions of Inconel 617
2. Variants of Alloy 800H
3. Hastelloy X and XR.

Conditions during irradiation, such as temperature, dose, dose rate, and material composition, determine the changes that will ultimately result. A substantial measure of understanding of radiation effects has been achieved, but investigation of these effects in the particular alloys being considered for NGNP applications will still be required under the particular conditions of interest. Additionally, microstructural stability under low-flux, long-time conditions at high temperatures will be evaluated for the candidate alloys. Initially, a detailed literature review of irradiation effects on all the potential candidate alloys will be conducted and design of the irradiation experiments will be started. Also, preliminary evaluations will be initiated regarding design of an irradiation facility for irradiation of candidate alloys under relatively low flux test reactor conditions, as well as identification and evaluation of potential test reactors.

RPV Component Irradiation Experiments

An irradiation facility to accommodate a relatively large complement of mechanical test specimens will be designed and fabricated for placement in a test reactor. The facility will, of course, include temperature control to allow for irradiation at the temperatures of interest and operate at a flux low enough to provide results that are applicable to the dose rates anticipated in service in the NGNP. Although the operating temperature of the RPV/CV may change with evolution of the design, it is currently planned to irradiate mechanical test specimens at 550 and 650ºC. The choice of these temperatures is based on the assumptions that (1) 650ºC is the highest possible operating temperature that can be envisaged for the RPV/CV at this time, (2) 550ºC is the lowest operating temperature that would allow for reasonable achievement of the objectives for the NGNP, and (3) those two temperatures would likely provide sufficient information for design and operation of the RPV at any intermediate temperature with respect to irradiation effects. A limited number of small test specimens (miniature tensile and Charpy impact specimens) of the potential candidate materials will be prepared for irradiation at high fluxes as an early screening experiment to identify alloys that may be particularly and unexpectedly sensitive to irradiation effects.

Additional irradiations of the preliminary candidate materials, both base metals and weldments, will begin later, with the choice of materials to be based on results of the literature review, as well as the baseline and aging tests completed at the time. For purposes of this plan, specimens to be irradiated will include those for tensile, hardness, creep and stress rupture, Charpy impact, fracture toughness, and fatigue crack growth testing. Based on the currently estimated maximum exposure of about $3x10^{19}$ n/cm^2 (>0.1 MeV) and 0.0075 dpa, the specimens will be irradiated to an exposure about 50% greater to accommodate uncertainties in the exposure estimates. A limited number of irradiated specimens will be aged in the impure helium environment for up to 10,000 h, tested, and examined by light optical and electron microscopy.

A decision to conduct test reactor irradiations beyond that time will be based on the results, but may include additional and more comprehensive irradiations of the final selected RPV/CV materials. However, as currently required by 10 CFR 50, Appendix H, and for reasons of prudence, the NGNP should incorporate a surveillance program. The specific design of the surveillance program, to include the specimen complement, will be based on the results obtained from the test program discussed above, but will likely include, as a minimum, tensile, Charpy impact, fracture toughness, and creep specimens. Because the NGNP is a demonstration reactor, it is recommended that the surveillance program be more extensive than might be required by the regulatory authority, such that it could serve as a test bed for irradiation experiments of more advanced materials that may be developed as the NGNP operations progresses.

Reactor Internals Irradiation Experiments

As stated previously, the neutron fluences accumulated in the metallic core internal materials are expected to be low relative to the tolerances of the structural alloys. Nevertheless, some consideration of irradiated effects is thought to be prudent. In the first year, a review of the radiation effects on the metallic reactor internal components will be undertaken. The review will include a collection of data produced on austenitic alloys irradiated at high temperatures. This body of information will be characterized in terms of materials, exposure conditions, and testing conditions.

In the second year, data judged to be pertinent to the NGNP will be evaluated in some detail and provided to the modeling activities underway in the task areas. Also in the second year, consideration will be given to irradiation exposures of candidate metallic internals materials with a high thermal to fast flux ratio. The selection of materials and exposure conditions will be undertaken. Working in collaboration with the task on the RPV materials, the design of experiments will be undertaken. In the third year, fabrication of the irradiation capsules will be undertaken. It is anticipated that the irradiation and post-test evaluations will be undertaken in collaboration with other tasks on the NGNP materials project. The budget does not include the costs for actual exposures.

VIII.3.1.2 NGNP Materials for High-Temperature Service

Given the accelerated materials qualification activities mandated by the early deployment of the NGNP, it will be necessary to rely almost completely on current commercial materials for the demonstration plant. However, to help optimize performance and minimize costs of follow-on NGNPs, as well as the other Gen IV reactors with later anticipated deployment dates, evaluation of advanced materials will also be initiated from the onset of the NGNP materials qualification efforts.

Four classes of advanced ferritic/martensitic steels, 9Cr-1MoVNb, 7-9Cr2WV, 3Cr-3WV, and 12Cr-1MoWV, are considered for PRV and CV. Additional two candidate materials are also selected as fallback options: the 2.25Cr-1Mo for a lower temperature operation, and austenitic stainless steel for their superior oxidation and corrosion resistance at significantly higher cost. At present, all the ferritic/martensitic steel in the ASME Boiler and Pressure Vessel Code have their limits at or lower than the desired PRV operating temperature of 650ºC. These candidate materials could also be considered for PCV. Due to the lower operating temperatures of PCV, however, the current LWR pressure vessel materials, SA508 grade 3 class 1 forgings or SA533 grade B class 1 plates, are considered for their lower cost. In addition, the 2.25Cr-1Mo is also considered for PCV dependent on the dissimilar materials welding issues between CV and PCV. Alloy 718, approved up to 566ºC in ASME Section III, Subsection NH, and types 304 and 316 stainless steels, with allowable stress intensities for bolting up to 704ºC, are considered for high-temperature closure bolting. It is necessary to increase their allowable temperature to that required for the NGNP and evaluate the need for inclusion of bolting in the irradiation program.

Stainless steels, high alloys, nickel base superalloys, and special advanced materials are considered for metallic reactor internals. Leading candidates include 316FR, 800H, 617, XR, 214 and ODS alloys. Cladding is also an option. Although data exist for some of these candidates for temperatures up to 1100°C, allowable temperatures for those currently Code approved materials are limited up to 980°C, while the internal components are expected to serve at temperatures up to 1000°C with an accidental excursion of 1200°C. The fatigue, thermal-fatigue, seismic, and other loadings that could produce damage are largely unidentified at present. Helium compatibility, radiation and thermal aging effects on properties, fabrication and joining are also issues that should be addressed. Metallic core support structures must conform to ASME Section III, Division 1, Subsection NG, and other core internals may conform to different rules. The SCS tubes may be limited to materials listed in ASME Section II, Part D, Tables 2A, 2B and 4. Candidate materials for metallic core components are also considered for piping, valves, recuperator, hot duct, CV bellow, butterfly valve, and SCS helium circulator ducting etc. Dependent on specific components, qualification issues include thermal fatigue, environmental effects, sigma phase embrittlement, creep, environmental and aging-induced degradation.

Candidate materials for IHX and the hydrogen plant HX are 617, X, and XR for temperatures up to 1000°C. Other nickel base alloys such as 740 and 230 are also considered. Qualification issues include transient thermal loading, degradation from helium impurities, aging effects, welding and fabrication. Other concerns may arise dependent on alternate IHX and HX designs.

Also included for qualification are high alloys, cast or wrought nickel base alloys for PCS components including turbine inlet shroud, turbine blades and disks, which are exposed to temperatures up to 1000°C. Some of these components require shorter design lives since they can be replaced at the 7-year maintenance intervals.

As the first step for qualification, a comprehensive and detailed review of the candidate materials will be conducted. The existing database for those alloys will be assembled, analyzed, and evaluated with respect to the design and operating requirements. Principal topics for review will include: fatigue, creep, creep-fatigue and stress corrosion cracking for lifetime prediction, impurity effects on degradation, aging effects, austenitic alloy sensitization, carburization, decarburization and oxidation, fabrication, effect of thickness on mechanical and fracture properties, high-temperature strength, stability, and long-time performance under irradiation. Based on the materials review, detailed research enabling the inclusion of the candidate materials into the ASME Code for the materials of construction will be defined and performed. For Code approved materials, additional testing may be required to extend temperature coverage but not nearly as much as that required for materials not currently approved for Code use. The extension of databases and ASME Codification will be developed and closely coordinated with the high-temperature design methodology activities. Dependent on specific component, additional databases will be required and developed for codification purpose.

VIII.3.1.3 Development of High-Temperature Design Methodology for NGNP

Of the several Gen IV reactor concepts, NGNP has the highest operating temperatures and, consequently, the most challenging high-temperature design methodology development needs. It also has, because of its near-term priority, the most demanding schedule. Simplified methods and interim design criteria must be in place prior to preliminary design.

The primary role of the High-Temperature Structural Design Methodology Task is two fold. First, it will provide the models required by ASME Code groups to formulate time-dependent failure criteria that will assure adequate NGNP component life. Second, it will provide the experimentally-based material models that are the bases for the inelastic design analyses required by Subsection NH of Section III of the ASME Boiler and Pressure Vessel Code, which governs the design and construction of elevated-temperature Class 1 nuclear components. Since it is not practicable to perform inelastic analyses of every region of every component and loading condition, the Code provides simplified methods for use in non critical regions. These simplified methods are also used extensively in preliminary design.

Unfortunately, the projected NGNP operating temperatures are well above the temperatures permitted for the current limited number of materials in Subsection NH. Only the temperature limits for Alloy 800 come close to those required for the NGNP vessels. Coverage for none of the materials is adequate for the very-high-temperature NGNP components. Furthermore, the allowable design life permitted for current Subsection NH materials is 34 years, in contrast to the 60 year life required for NGNP components.

Thus, NGNP will require that new materials be incorporated into Subsection NH. Furthermore, because these materials will operate at the upper end of their useful elevated temperature range, they will exhibit some basic behavioral features drastically different from those on which many of the current Code rules and simplified methods are based. Thus, the new materials will also require the development of new or modified rules and simplified methods. The needs associated with developing and applying very-high-temperature criteria were itemized in the 1980’s when a draft nuclear Code case was developed for the use of Alloy 617 for relatively short times at temperatures up to 982°C.

The strategy for the high-temperature design methodology development effort for NGNP is to focus initial efforts on a single medium high-temperature vessel material (modified 9Cr-1Mo steel, Grade 92) and a single very-high-temperature component material (nickel-base Alloy 617). Work on these alloys will provide an initial focus for Code development work, and the resulting criteria and design analysis and assessment methods will provide the framework and springboard for expeditiously addressing additional down-selected NGNP materials as they are identified. The initial effort will also provide the near-term criteria and tools needed by NGNP designers to develop conceptual and preliminary designs.

VIII.3.1.4 NGNP Materials Compatibility

<u>Helium Environment</u>

From a compatibility view point, the internals of reactor will operate in a helium environment, and the externals of the pressure vessel will operate in air. The reactor will operate at 1000%C and the pressure vessel will operate at 650%C. The interactions between structural materials in high purity helium atmospheres associated gas cooled reactors have been the subject of numerous investigations. The results of these studies conducted by various organizations in USA, Germany, England, Norway, Japan, and other places have demonstrated the importance of small changes in impurity levels, high temperatures and high gas flow rates. Metallic materials can be carburized or decarburized, and oxidized internal or at the surface. These corrosion reactions, depending on the rate, can affect long term mechanical properties such as fracture toughness.

The composition of helium cooled reactor must be defined and bounded. Because the low partial pressures of the impurities, the oxidation/carburization potentials at the metallic surface of a gas mixture is established by the kinetics of the individual impurity catalyzed reactions at the surface. The main impurities associated with helium cooled reactors are H_2, H_2O, CO and CH_4. The hot graphite core is considered as reacting with all free O_2 and much of the CO_2 to form CO, and with H_2O to form CO and H_2. In addition, in cooler regions of the core, H_2 reacts with the graphite via radiolysis to produce CH_4. Because of the change in surface temperatures around the reactor, and associated changes in reaction mechanisms and rates of reaction on bare metal versus on scaled surfaces, reaction rates and order of reactions are important.

The overall stability of the proposed helium environment must be evaluated in order to ensure that testings proposed in various sections of the program are performed in environments that have consistent chemical potentials. In addition, the corrosion of metals and nonmetals will be evaluated to establish baseline data where it does not exist. These tests will be performed at temperatures to include at least 50%C above the proposed operating temperature.

Emissivity

The external air environment is significant in that the pressure vessel must be able to radiate heat at 650%C throughout the life (60 years) of the reactor. It is therefore necessary to have a stable, high-emissivity surface on the pressure vessel material such as, 9Cr-1MoVNb and variants, at elevated temperatures. While, the emissivities of steel can be increased by the formation of an oxide film, the conditions under which this film can be created and the stability of this film in air (including the effect of humidity) at operating temperature needs to be established. An industrial partner will be used to provide for scaling of the materials and methods that have proven to be viable.

VIII.3.1.5 Graphite Materials

Graphite will be used as a structural material and neutron moderator for the NGNP core, and as the permanent side reflectors and for the core support structure. A significant challenge related to graphite is for the NGNP is that the previous U.S. standard graphite grade qualified for nuclear service, H-451, is no longer commercially available. The precursors from which H-451 graphite was made no longer exist. Hence, it will be necessary to qualify new grades of graphite for use in the NGNP. Fortunately, likely potential candidates currently exist, including fine grained isotropic, molded or isostatically pressed, high-strength graphite suitable for core support structures, fuel elements and replaceable reactor components, as well as near isotropic, extruded, nuclear graphite suitable for the above-mentioned structures and for the large permanent reflector components.

Technical Issues

Near-isotropic, extruded, nuclear graphite’s (e.g., grade H-451 manufactured by SGL Carbon) were developed in the 1970’s for large helium cooled reactors such as the Fort St. Vrain reactor. However, grade H-451 graphite has not been manufactured in the United States for more than 25 years. Consequently, an assessment of available alternatives graphites must be made.

New near isotropic, extruded, nuclear graphite have been developed in the United States and Europe for the South African Pebble Bed Modular Reactor (PBMR). The new (currently available) graphites are GrafTek (UCAR) grade PCEA—a petroleum coke graphite, and SGL Grade NBG-10—a pitch coke graphite based on United Kingdom Advanced Gas-Cooled Reactor (AGR) fuel sleeve graphite. These graphites are candidates for the fuel elements and replaceable reactor components of the NGNP. The fine-grained isotropic, molded or isostatically pressed, high-strength graphite suitable for core support structure includes Carbone USA grade 2020 and Toyo Tanso grade IG-110. Toyo Tanso grade IG-110 was used in the Japanese HTTR for fuel blocks and in the Chinese HTR-10 pebble bed reactor. These fine-grained materials are also suitable for the fuel elements and replaceable reactor components of the NGNP. Graphite's suitable for the large permanent reflector components are currently in production (e.g., SGL grade HLM or GrafTek [UCAR] grade PGX). Some data are available for these graphite grades. Grade PGX was used for the permanent reflector of the Japanese HTTR, also PGX and HLM were used in Fort St. Vrain for the core support and permanent reflectors respectively. Fine-grain, high strength, graphite's are available from POCO Graphite, Inc. However, the available billet sizes are small, thus not suited for NGNP core applications.

A materials properties design database must be developed for the selected NGNP graphites, including data for the effects of reactor environment on properties (including neutron irradiation and irradiation creep).

Graphite, Reflectors, and Supports Qualification Test. Although candidate graphite materials are known, certain tests must be conducted to verify the candidate's relevant material properties meet the claims of the manufacturer. The Preliminary Selection process will need limited irradiation response data for the different grades of graphite. These test results will be used to establish the general behavior of a particular grade of graphite and confirm that it behaves similarly to previously "qualified" (for other nuclear reactors), near-isotropic, nuclear grades of graphite.

The grade of nuclear graphite (H-451) previously used in the United States is no longer available. New graphite grades have been developed and are currently considered as candidates for the NGNP. Early in the program, it will be necessary to review and document the existing data, from all available sources, on the properties of these new grades of graphite. Irradiation data from ongoing experiments in the Petten Reactor (European program) will be of great value. A complete properties database on the new (available) candidate grades of graphite must be developed to support the design of graphite core components. Data is required for the physical, mechanical (including radiation induced creep) and oxidation properties of graphites. Moreover, the data must be statistically sound and consider in-billet, between billets, and lot-to-lot variations of properties. The data will be needed to update and benchmark existing design models for graphite performance. Since the available near-isotropic, extruded graphites are somewhat similar to the prior grade H-451, design models for H-451 can be incrementally adjusted for the currently available graphites as new data becomes available. This review will provide data that will be input into the preliminary selection process.

As part of the preliminary selection process, a radiation effects database must be developed for the currently available graphite materials. As mentioned above, there is the potential to leverage data from European Union activities in the area of irradiation experiments on PBMR graphites (Petten Reactor irradiation experiments are currently being initiated). However, it is anticipated that a substantial number of additional graphite irradiation tests will be needed to complete the database. Since NGNP graphite service temperatures are anticipated to be as much as 200°C greater than that in the GT-MHR, additional data are required for all properties at these higher temperatures, including radiation damage effects. Therefore, in order to be qualified for the NGNP, existing graphite behavior models need to be modified based on sound materials physics and then validated/verified against new data for the currently available graphite grades. Property data must support the service conditions, including effects of higher temperature, helium gas (plus air and water), and neutron irradiation effects. Irradiation creep data for the candidate graphites must also be obtained.

Graphite Baseline Materials Test Program

The baseline graphite test program will "fill in the blanks" in the database that cannot be abstracted from European and Japanese programs. The baseline materials test program must be sufficient to fully characterize and quantify property variations within candidate graphite billets arising from the raw materials forming process (e.g., parallel and perpendicular to the forming axis), as well as spatial variations (i.e., billet edge and center). Microstructural characterization of candidate graphites will be conducted in order to establish filler particle and pore size distribution (required for fracture modeling). X-ray diffraction (XRD) will be applied to establish crystal parameters and appropriate crystalinity factors for neutron irradiation behavior modeling and prediction.

Prior work and data for nuclear graphite behavior will be reviewed and assessed in an effort to minimize the extent of the testing program.

Physical and Mechanical properties to be determined include:

- Mechanical Properties – Strength (tensile, compressive, flexural), Biaxial/multiaxial strength, Stain to Failure, Elastic Modulus, Poisson's Ratio, Fatigue Strength, Fracture Toughness.
- Thermal Properties – Thermal Conductivity, Thermal Diffusivity, CTE, Emissivity, Specific Heat.
- Tribology – Significant work has been previously performed on graphite-graphite friction couples. This work needs to be reviewed and documented in the graphite materials database. Previous work did not indicate a significant problem with regard to erosion or grinding of components other than dust generation (which needs to be quantified- including measurements made during FSV reactor operation). The graphite dust will have to be removed using the helium purification system. System requirements can be developed from the dust generation data. Previously, circulating graphite dust was not found to be an erosion or grinding problem, but graphite dust will need to be cleaned out of the helium to prevent clogging of small instrumentation passages, etc.

The chemical purity and Boron equivalent content of the candidate graphite will be determined.

Unirradiated Graphite Material Qualification Program

For qualification, variations between billets within a single lot and between different lots must be fully characterized. Sufficient data must be taken such that the data are statistically significant. Therefore, it will be necessary to purchase full-size billets of graphite from several lots for qualification testing. Ideally, several graphite billets will be purchased from a pre-production lot (this may require the purchase of the entire lot) in order to establish and quantify the extent of in-billet and between billet property variations. Moreover, graphite purchased should meet the requirements established for the ASTM Nuclear Graphite Materials Specification. Graphite thus purchased will additionally be used for ASTM test method development.

For qualification, property data is needed as a function of temperature and environment (helium). Moreover, the long-term effects of impurities in the coolant helium (air, water oxygen) on the graphite properties must be established (Graphite oxidation). All of the properties determined under the baseline graphite materials test program will need to be re-assessed for the effects of oxidation from helium coolant impurities (air, CO_2, water). Graphite air oxidation kinetic data must be obtained for the candidate graphites for air-ingress accident simulation and modeling.

Design specification data will be required on the helium coolant purity limits, as this will control the severity of the property degradations.

Graphite Irradiated Materials Test Program

Significant structural changes occur upon neutron irradiation. The single crystal effects and gross structural effects combine to modify practically all of the properties. Thus, for preliminary selection of candidate graphites, those properties listed in the baseline program above must be examined for the effects of irradiation at a temperature representative of service conditions.

The effects of neutron irradiation over the temperature and dose range appropriate to the NGNP must be established as part of the qualification process. A significant body of data on the effects of irradiation exists and prior data on near-isotropic nuclear graphites will serve to guide experimentation. The selected reactor type (e.g., prismatic or pebble-bed) will significantly impact the irradiation program because the anticipated maximum neutron doses are much greater in the pebble-bed design. The irradiation experiment temperatures should bracket the design envelope for the selected reactor.

A major component of the irradiation Qualification-testing program will be determination of the irradiation creep coefficient in compression and tension. Under irradiation, graphite will creep at temperatures where thermal creep does not occur. The rate of creep is related to the applied stress, the initial modulus and neutron dose and temperature.

Codes and Standards

Significant activity is required to bring the existing graphite codes and standards to an acceptable condition. The proposed section III Division 2, Subsection CE of the ASME B&PV Code (Design requirements for Graphite Core Supports) was issued for review and comment in1992 and no action has been taken on this code since that date. There is activity underway currently (funded by the NRC) to reinitiate the “CE” code committee and begin the process of code case approval. However, significant revision of the code is required as well as expansion of the code to the higher temperatures envisioned for the

NGNP. Moreover, the NRC has indicated that the code should be revised to increase the neutron dose limits to levels appropriate to the PBMR.

Graphite test standards have been developed for nuclear grade graphites (ASTM C-781: Standard Practice for Testing Graphite and Boronated Graphite Components for High Temperature Gas-Cooled Nuclear Reactors). This ASTM standard must be further expanded to cover required test methods including, Fracture Toughness, XRD, Graphite Air Oxidation, Boron Equivalency. This activity is being led by ORNL. Moreover, the standard must address specimen size issues as they relate to the preparation of graphite irradiation specimens. ASTM is currently preparing a nuclear grade graphite material specification under the jurisdiction of Committee DO2-F.

<u>Fabrication Infrastructure Development Requirements and Program</u>
Appropriate NDE methods must be developed for large graphite billets and components. Such methods must be applied prior to accepting production billets for fuel element/component machining and will be useful for subsequent in-service inspection.

VIII.3.1.6 Ceramic And Composite Materials For NGNP

The very high temperatures that will exist within the primary circuit during both normal and off-normal conditions will require the use of non-metallic materials, in addition to graphite, where temperatures exceed those at which metallic materials can operate. Primary applications for these are for structural and insulation materials for use within the reactor vessel, piping, heat exchangers, and turbo-machinery. Primary structural material candidates for these applications are carbon-carbon composites and possibly SiC-SiC composites where very high radiation exposure resistance is required, such as control rods. Some illustrative components that could be manufactured from carbon-carbon composite material components of the NGNP are listed below:

- Control rod structural elements
- Control rod guide tubes
- Hot duct insulation cover sheets
- Lower core support insulation blocks
- Upper core restraint structure blocks
- Upper shroud insulation cover sheets
- Shutdown Cooling System entrance insulation.

Carbon-carbon composite technology development has been brought to a very mature state by the aerospace industry and there are several manufacturers of carbon-carbon composites of the type that are thought to be required for NGNP components. However, they have not qualified any of their recent high-performance materials for nuclear applications. Potential candidate structural composite materials may need modification and will need to have the suitability of mechanical and thermo-physical properties for reactor applications confirmed in the as-fabricated, aged, and irradiated conditions. In addition, the manufacturers and their prime candidate materials must be examined for repeatability, quality, and eventual size of manufacture, as many of the parts will be very large.

Large-sized SiC/SiC composites are not as available as C-C composites, and hence they are not suitable for larger NGNP components. However, it is recognized that these composite materials have undergone rapid development within the last ten years. The result is a very limited database for the newest, radiation resistant materials. Even though existing data has shown that C-C composites should easily withstand the neutron doses in all NGNP components, except of the control rods, data shows that carbon-carbon composite control rods will likely need to be replaced, whereas SiC/SiC composite rods would likely survey the full 60-year life of the NGNP. To assess the technical and economic viability of using SiC/SiC control rods, a comparative irradiation study is planned to compare the differences between silicon carbide composites and carbon-carbon composites.

NGNP insulation will include both structural ceramics of low thermal conductivity (typically designed to be stressed in compression, since ceramics exhibit high compressive yield strengths) and low-density ceramics (e.g., foams or fibers) that will provide excellent thermal insulation. There are many design concepts available to achieve insulation. For example, a meter of graphite (K_{th}> 10 W/m-K) thickness plus 0.2 meter of carbon-carbon composite blocks is sufficient to insulate the lower metallic core support structure from the core outlet gas. However, where room is limited to a few inches of insulation thickness to do the same job, a more efficient form of insulation is required. Structural ceramics, such as high-purity alumina and stoichiometric mullite would likely be used a monoliths because of their high density and creep resistance Where more room is available and significant structural support is not required, a suitable insulation system, is to sandwich Al_2O_3 - SiO_2 mixed ceramic fiber mats between metallic or C-C composites cover plates, depending upon temperature requirements, that are attached to the primary structure. An alternative to the sandwich design is one where ceramic fiber "blankets" of various configurations can be attached to cooler outside structures using refractory pins and washers.

Operating conditions for fibrous insulation include low neutron fluence and gamma flux, and high temperatures. Mechanical loads on the thermal insulation result from differential thermal expansion, acoustic vibration, seismic vibration, fluid flow friction, and system pressure changes. Qualification of the mechanical and thermo-physical properties of the insulation materials will be required to ensure they do not excessively degrade due to aging, irradiation, or exposure to the reactor coolant over time.

VIII.3.2 GFR Reactor-Specific Materials

The GFR system offer a closed fuel cycle through high conversion or breeding of fissile materials. GFRs using a direct Brayton cycle have the potential to combine the advantages of high sustainability and economic competitiveness, while making nuclear energy benefit from the most efficient conversion technology available. The reference concept is a 600 MWth/288 MWe, helium-cooled reactor system operating with an inlet temperature of about 490°C and an outlet temperature of about 850°C, and using a direct Brayton cycle gas turbine.

Temperatures of core structures will range from 500-1200°C under normal operation and may climb to 1650°C under accident conditions. Maximum neutron exposure of core components will likely be up to 80 dpa or higher. The bulk of the out-of-core structures will experience operating temperatures of 440-850°C. The vessel is expected to operate in the temperature range of 440-600°C for normal operation, with excursions to 650°C for up to 50 hours. It is expected that the fluence on the vessel can be limited to about $1x10^{18}/cm^2$ (E>0.1 MeV) with adequate shielding within the RPV.

At the current time, additional details regarding system and component design are being initiated. Updates of this document will provide better definition of specific component conditions and materials operating requirements as they become available. In the meantime, it is reasonable to assume that, with a few significant exceptions, the materials needs for the GRF will be enveloped by those for the NGNP. The greatest difference will be the requirements for core materials. In the GFR, the use of graphite as a structural material will not be possible due to its strong moderation (softening) of the fast spectrum inherently needed by the GFR. In addition to very high temperatures, the core materials will experience accumulate much higher radiation damage than a thermal reactor. This combination of requirements will require the use of alternative high-temperature core materials. Ceramics (carbides, nitrides, oxides, silicides) have the greatest potential for use in the GFR core but, to prove their feasibility, significant R & D will be needed because the starting background of knowledge needed for nuclear applications is very poor on these materials for fast reactor core applications. As back-up, refractory metallic alloys (W, Nb, Ta, Mo basis) will need to be evaluated but it will be necessary to quantify the extent to which, the use of these materials will be limited by neutronic criteria.

Other significant difference in materials requirements from the NGNP will arise from the lower 850°C outlet temperature of GFR, and the possibility of using a supercritical CO_2 Brayton cycle for electric power generation. Compatibility of primary circuit materials with impure helium, especially in the absence of graphite, and with actinide compounds must be evaluated. In addition, a key material issue for a helium-to-supercritical CO_2 intermediate heat exchanger and other secondary components is corrosion.

Limited materials R&D activities that have already been identified are included within section VIII.8 on materials milestones. Once the initial materials requirements and R&D planning documents are completed (scheduled for this current fiscal year), more extensive GFR-specific materials R&D plans will be incorporated into subsequent updates of this document.

VIII.3.3 SCWR Reactor-Specific Materials

VIII.3.3.1 Materials for SCWR Radiation Service

Factors that will determine the service life of materials for the SCWR are a combination of corrosion in supercritical water and radiation effects. The non-fuel materials of the reactor that are expected to experience significant neutron displacement doses are: (1) core structural materials, (2) core support structures, and (3) pressure vessel. In the first category are the fuel cladding, fuel rod spacers (spacer grid or wire wrap), water rod boxes, fuel assembly ducts, and control rod guide thimbles. The second category includes control rod guide tubes, upper guide support plate (UGS), upper core support plate (UCS), lower core plate (LCP), calandria tubes, core former, core barrel, and threaded structural fasteners. The reactor pressure vessel (RPV) includes two low temperature inlet nozzles and two high temperature outlet nozzles. Insulation materials will also be needed for the reactor internals that separate the hot outlet coolant from the inlet coolant, and for the pressure vessel outlet nozzles.

The reactor will operate at a pressure of 25 MPa, above the thermodynamic critical point of water. The above components will be exposed to supercritical water, ranging from the low temperature inlet at 280 ºC up to the outlet slightly higher than 500 ºC. The coolant changes from a compressed liquid to a fluid nearly an order of magnitude less dense than ordinary water in traversing the core from bottom to top. Doses vary over a wide range, from hundredths of a dpa for the RPV, UGS, UCS, LCP, and calandria tubes to 15-20 dpa for the replaceable fuel assemblies and core former. Under normal operation the highest temperatures of up to 620ºC will be experienced in the upper part of the core by the fuel cladding, fuel rod spacers, and the core former. At the same time the bottom of the core will be at a temperature of 280 ºC. Under off-normal conditions the fuel cladding temperature could reach 840 ºC.

Materials qualification will be carried out as a progressive program of selection from a range of candidates mainly in the Fe-Ni-Cr alloy system, then screening of materials by testing to select promising candidates, followed by alloy modification where necessary for specific conditions, and alloy development in the event that satisfactory alloys cannot be obtained in the earlier stages. The range of compositions within the Fe-Cr-Ni alloy system within which alloys with acceptable mechanical behavior and dimensional stability currently exist, or could be developed, may be divided into four broad categories namely, a) austenitic stainless steels, b) ferritic and ferritic-martensitic steels, c) high alloys (Fe < 50 wt.%) and d) Ni-based alloys.

Other materials are also included. For example, for control rod thimbles experiencing temperatures < 300 ºC, zirconium alloys are candidates based on their proven performance in currently operating reactors. Consideration also will be given to the potential application of ceramic materials such as silicon carbide composite materials. These materials have been developed primarily for applications requiring high strength at temperatures well above those of the SCWR. Although nothing is known regarding their behavior in SC water conditions, such materials could offer significant advantages over metallic in some cases. Where the application requires it, the outer composite layer could be fabricated with a higher porosity to act as an insulator.

There is insufficient knowledge at present regarding the behavior in supercritical water of the materials described above to rank them in terms of irradiation-assisted stress corrosion cracking (IASCC). Within each category, there exist numerous compositions that have the basic strength and ductility properties to meet the operating requirements of the SCWR. For the reactor vessel, with an operating temperature and irradiation

exposure similar to that of current generation pressurized water reactors (PWR), the primary candidate materials for the RPV shell are those currently used in PWRs, namely variants of SA 508 steel. However, because of the high pressure of 25 MPa, a vessel of this material would have to be about 50 % greater in wall thickness than current practice. Therefore consideration will also be given to higher strength chromium steels containing solution strengtheners in order to reduce the section thickness.
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The materials program consists of two overlapping activities: a) research and development to define prime candidate alloys, and b) a materials engineering design data effort. The former entails a sequenced set of testing and performance evaluation stages in which an initial set of potential candidate materials is reduced to a limited number of prime candidates through testing in increasingly complex and aggressive environments. Throughout the R&D program, it will be essential to adopt an integrated theoretical modeling and experimental approach in order to build the scientific knowledge needed to understand the mechanisms controlling behavior and to provide a rational basis for developing improved alloys. R&D will ensure the viability of the SCWR. It will yield alloy compositions and thermo-mechanical treatments with demonstrated capability to meet the intended service conditions. The second activity involves extensive evaluation and qualification of the prime candidates to develop a materials engineering design database that meets licensing requirements. The product of this phase will be specifications for producing materials in the required product forms, an approved data base on properties, the structural assessment methods required to support design, construction, and licensing, and a reliable basis for the prediction of materials performance throughout the expected lifetime.

The behavior of alloys in supercritical water absent irradiation will be the dominant feature of the initial phases of the R&D program. In the following stages of the program, irradiations of selected materials will be carried out, culminating in irradiations of the best performing materials in irradiation facilities containing supercritical water. The approach will develop information on the broad response of the four alloy categories, as well as on the silicon carbide composites, and on the effects of specific compositional and microstructural variations within these classes.

Selection of alloy compositions and conditions for the initial evaluations in supercritical water will be guided by existing data in three different areas. Firstly, materials will be included for which there is substantial information on behavior in current water reactors. These benchmark materials provide a basis for identifying acceleration of known phenomena, or for detecting the development of new phenomena, in supercritical conditions. A second source of information to be considered is the experience derived from the operation with a variety of materials in fossil fired supercritical steam power plants. The third basis for alloy selection is the vast body of data on the effects of neutron displacement damage on materials, which has been developed over the past 30 years of LWR, fast breeder reactor, fusion power and basic science programs worldwide. This database will provide a rationale for the exclusion of alloys based upon well-documented behavior in terms of radiation embrittlement and dimensional instability under the conditions of temperature, mechanical loading and neutron dose projected for the core internals. The work will be carried out in a coordinated program utilizing existing experimental facilities at various U.S. institutions in close collaboration with similar international efforts.

VIII.3.3.2 SCWR Materials Compatibility

The mechanisms for environmentally sensitive cracking in water-cooled reactors that have been observed include intergranular stress corrosion cracking (IGCC), irradiation-assisted stress corrosion cracking (IASCC), and corrosion fatigue. These mechanisms are affect by several variables including metallurgical structure, irradiation inducted grain boundary segregation, and oxidizers/reducers in the aqueous environment.

There are several aspects of the water chemistry of the SCWR that will impact the corrosion behavior of materials of construction. The concentrations of the transient and stable species due to radiolysis of the water at the higher operating temperature (as compared to LWRs) may well be significantly different. The chemical potential of oxygen and hydrogen peroxide, which will be significantly different in the supercritical fluid, will affect the corrosion potential of the water. This in turn determines whether magnetite (Fe_3O_4) or hematite (Fe_2O_3) forms and the morphology of these films, which are important to corrosion control on low alloy steels. Note that the low alloy pressure vessel steel will generally not be exposed to an aqueous environment due to the stainless steel weld overlay cladding, however, possible contact of the pressure vessel steel with the supercritical water will need to be quantified in the safety assessment.

The chemical potential of the hydrogen should change as much as the chemical potential of the oxygen and hydrogen water chemistry may be just as effective in reducing the oxygen content. However, a decrease in the critical reaction rate of the OH radical with hydrogen above
300 %C has been observed. Because the radiolysis in the core is kinetically controlled, it might require much more hydrogen to suppress the oxygen and peroxide generation. If too much is required, metal hydriding could occur. The trade-off between these effects, will, to a large extent, determine how much of the LWR and fossil plant water chemistry control experience is applicable to the SCWR. The control of pH, while theoretically possible, may be difficult in practice, especially in the 300 to 500 %C temperature range. The pH of the water is important in setting the corrosion potential and rate, and to some extent, the mode of corrosion. A range of pH has been successfully employed in LWRs, and this approach will need to be explored.

VIII.3.4 LFR Reactor-Specific Materials

LFR systems are Pb or Pb-Bi alloy-cooled reactors with a fast-neutron spectrum and closed fuel cycle. Options include a wide range of plant ratings, including a long-refueling-interval transportable system ranging from 50–150 MWe, a modular system from 300–400 MWe, and a large monolithic plant at 1200 MWe. These options also provide a range of energy products. The focus of the U.S. program is on transportable concepts that are small factory-built turnkey plants operating on a closed fuel cycle with very long refueling interval (15 to 20 years or longer) cassette core or replaceable reactor module.

Near-term systems are limited by material performance to outlet temperatures of about 550°C. Both Pb and Pb-Bi are coolant options for this reactor. Pb having probable material corrosion improvements, but limiting core ΔT, and Pb-Bi providing more temperature flexibility but raising issues of Po-210 and Bi corrosion. The favorable properties of Pb coolant and nitride fuel, combined with development of high temperature structural materials, may extend the reactor coolant outlet temperature into the 750–800ºC range in the long term, which is potentially suitable for hydrogen manufacture and other process heat applications. In this option, the Bi-alloying agent is eliminated. The required R&D is more extensive than that required for the 550ºC options because the higher reactor outlet temperature requires new structural materials, coolant technology and nitride fuel development.

At the current time, additional details regarding system and component design are being defined. Updates of this document will provide better definition of specific component conditions and resulting materials operating requirements as they become available.

VIII.3.4.1 General Considerations for LFR Materials Research

Three primary factors will most affect the properties and choice of the structural materials from which the LFR components will be fabricated. These are effects of irradiation, high-temperature exposure, and interactions with molten lead or lead-bismuth coolants to which materials in the primary circuit are exposed. An extensive testing and evaluation program will be required to assess the effects that these factors have on the properties of the potential materials for LFR construction to enable a preliminary selection of the most promising materials to be made and to then qualify those selected for the service conditions required. Structural materials needs for LFR systems can be divided into five general classes, those for: cladding, reactor vessel, internals, heat exchangers, and balance of plant. These are addresses in subsequent sections.

Two of the three primary considerations for LFR service, irradiation and high-temperature exposure will largely be addressed with the research planned for crosscutting materials those for the NGNP. While the levels of neutron exposure for the LFR will be quite high (up to 200dpa) for the metallic components, most of the same mechanisms identified at lower fluences will still be of concern, though at a much greater level. Irradiation-induced swelling of structural alloys at the very high fluences anticipated for LFR internal components will be a much greater limitation for selection and operation of metallic materials. The third primary consideration, materials interactions with molten lead or lead-bismuth coolants is unique to the LFR and described below.

VIII.3.4.2 Materials Compatibility and Corrosion with Pb and Pb-Coolants in LFRs

Materials compatibility concerns for structural metal alloys that are in contact with the coolants for the LFR will be very significant. General corrosion, thermal-gradient-induced mass transfer, and even stress corrosion cracking and liquid metal embrittlement are all potential failure mechanisms that must be addressed.

Most of the history understanding of structural metal in a Pb or Pb-Bi environment is derived from Russian programs, in which significant development was performed to understand and deploy materials and coolant chemistry control schemes for lead-alloy cooled systems. Outside of Russia, the technological readiness level of lead-alloy nuclear coolant technology is at a much earlier development stage, but the partial knowledge of the Russian experience available to the Western technical community has been factored into this materials plan.

Russian LBE nuclear coolant technology relies on active control of the oxygen thermodynamic activity in LBE to control corrosion and coolant contamination. Within this framework, a series of structural materials were developed and tested in Russia for enhanced corrosion resistance and acceptable lifetime for operating temperatures below 550^{o}C, with fuel cladding temperature below 650^{o}C. Unfortunately, the most advanced Russian alloys, although similar to some Western alloys, have no direct counterpart.

The oxygen control technique, when properly applied, leads to the formation of “self-healing” protective oxide films on the surfaces of the materials in contact with lead-alloys. This is because the base element (typically Fe) and alloying elements (Cr, Ni) of many structural materials have higher chemical affinity to oxygen than to the coolant alloy constituents. Without such protective measures, Fe, Cr and especially Ni all have non-negligible solubility in lead-alloys that causes severe dissolution attacks.

Oxygen sensors and control systems are thus important components of the reference coolant technology. Alloying materials with elements promoting tenacious and protective oxides (e.g. Si and Al), or treating/coating the surface with appropriate materials for enhanced corrosion resistance, have been developed and tested with oxygen control.

For materials used for operating conditions at the high end of the reference technology (above 500^{o}C), it is necessary in some cases to precondition them, i.e. pre-oxidize them so that the kinetics is favorable for growth of protective oxide film during operations. There has been little systematic evaluation and development in this area.

For promising candidate materials, especially the ferritic and martensitic steels for fuel cladding and other high temperature applications, preconditioning (e.g. hot dipping in oxygen saturated LBE bath) tests and subsequent corrosion testing in lead-alloys needs to be performed.

Using steels as the main structural materials, the existing LBE technology requires a proper control of the oxygen level to mitigate the steel corrosion problem. Under this framework, if oxygen is depleted, liquid metal corrosion via dissolution attack, and possibly liquid metal embrittlement, can occur. However, at high temperatures in Pb, oxidation kinetics may be accelerated too much and become detrimental. Within this higher temperature range, the mechanical properties of some refractory metals and alloys improve but oxidation problems compound (e.g. internal oxidation of Nb). So oxygen-free coolant technology may be needed for high temperature reactors.

It will also be very important to assess weight loss by corrosion. Temperature gradient mass transfer will likely be an important phenomenon in these systems and experiments should be designed specifically to investigate it. In a system with a temperature difference and with alloy constituents that are soluble in the coolant, it is possible to dissolve from the higher temperature regions and reprecipitate on cooler regions. Because there is a temperature gradient, equilibrium levels could never be established in the coolant, so there is an "engine" that unavoidably transfers mass from one part of the system to another. This would occur in addition to other forms of corrosion. In some liquid metal systems temperature gradient mass transfer has turned out to be the primary issue, even leading to complete blockages in some cases. Test loops with higher temperature and lower temperature sections and appropriate specimens in each region would be needed to assess this issue.

Recent development of lead-alloy spallation target and coolant technology worldwide for accelerator driven systems (ADS) has advanced the state of the art in the West considerably. There is now substantial amount of experimental evidence that the main features of the Russian lead-bismuth eutectic (LBE) nuclear coolant technology are valid for forced circulation in small to medium loop type systems. Corrosion tests by various international groups indicate that there are qualified structural materials (US, European and Japanese) for the temperature and flow conditions of the Russian reactors. However, to achieve the high potential aimed for in the advanced reactor system concepts, a significant amount of R&D is needed in the areas of materials and coolant chemistry control.

VIII.3.4.3 Materials for LFR Cladding and Core Internals

Cladding material for LFR systems must be compatible with metal or nitride fuel, corrosion resistant in lead or lead-bismuth coolants, and have adequate strength, ductility, toughness, and dimensional stability over the operating temperature range and to doses up to 200 dpa.

Because of the desire to operate to high dose, ferritic-martensitic (F-M) steels are the primary candidates for cladding in the lower temperature LFR. Because of the extensive work on HT9 for the earlier LMR program, for lower temperature (550°C outlet) LFR systems, HT9 is an initial reference cladding material. However, other more advanced F-M steels offer substantial strength and toughness advantages over HT9, and will probably perform better.

The corrosion resistance of F-M steel still needs to be proven before it is chosen as the cladding. Both Russian experience and preliminary U.S. corrosion studies indicate that elevated silicon levels may be required to provide adequate corrosion resistance when using oxygen control as the method for cladding corrosion protection. Additionally, earlier U.S. work has indicated that the formation of intermetallic or nitride surface layers based on Zr, Ti, and/or Al may provide satisfactory corrosion resistance. If alloys with higher silicon are required, the irradiation test base must be established for the new higher silicon alloys.

Some of the more advanced candidate materials include 9Cr steels T91 and 9Cr-2WVTa, the third-generation steels NF616 (a 9Cr-0.5Mo-1.8WVNB steel) and HCM12A (a 12Cr-0.5Mo-1.0WVNbN steel), should be given consideration. To obtain further significant improvements in high-temperature creep strength from ferritic steels, oxide dispersion-strengthened (ODS) steels will likely have to be produced and evaluated.

Qualification of any of these materials requires establishing both corrosion resistance and acceptable mechanical performance and dimensional stability. Corrosion testing of all of the ferritic-martensitic steels is important in increasing the potential operating temperature of LFR systems. A final possibility is to coat one of the steels in a manner that provides corrosion protection but maintains the acceptable mechanical and dimensional stability performance. Coating and surface modification technology is an important component of the cladding and core internals development program and will need to be evaluated, particularly for the higher desired operating temperatures.

For significantly higher temperature (800°C) applications, steels are not likely to be successful as cladding materials. For the higher temperature applications, ceramics, refractory metals, or coated refractories may be necessary. For these high-temperature candidates, the existing materials database comes from the fusion and space programs, but such data are limited and not sufficient for qualifying a material for reactor operation. Moreover, the consideration of such very high-temperature materials is based only on mechanical properties and dimensional stability metrics and does not consider corrosion

Based on the development work in the fusion programs and early promising results in lead corrosion tests, SiC and SiC composites would be primary candidates for 800°C application although high dose radiation resistance, cost, and fabricability are still major open issues. Tantalum alloys are also expected to be resistant to lead corrosion although they may not be adequate from a neutronics standpoint.

Materials considerations for core internals such as ducts, grid plates, core barrel, and other piping are similar to those for cladding, but may not have nearly as high a radiation exposure. Where the radiation dose is low, austenitic stainless steels could be considered.

VIII.3.4.4 Materials for LFR Reactor Vessel

The reactor vessel for an LFR must contain the lead coolant at primary inlet temperature. It also must be seismically qualified to hold the volume of lead during operation and shipping. Finally, it must have acceptable mechanical properties over the vessel lifetime (15-20 years). The LFR operates at atmospheric pressure.

For LFR vessels, if the neutron exposure is low enough to avoid swelling, austenitic stainless steels such as 316 or 304 are primary candidates. The EBR-II vessel, which operated in a similar temperature and pressure regime, was built of 304 stainless steel. For pool type designs, the vessel will be in contact with the coolant at the primary inlet temperature and corrosion resistance in low flow or stagnant lead alloy must be verified. HT9 and other ferritic-martensitic steels could also be considered.

The ASME codes have complete cases for only four alloys, 304, 316, 2 1/4 Cr-1 Mo steels, and Alloy 800H. The 304, 316, and Alloy 800H have maximum approved temperatures in the range of 760-816°C. Any other vessel materials or other temperatures will require additional data.

The materials selection, development, and qualification requirements for the vessel are very similar to that for cladding and core internals. Corrosion resistance must be confirmed for 304, 316, and possibly Alloy 800H. If these are inadequate, alternate materials or approaches must be established.

VIII.3.4.5 Materials for LFR Heat Exchangers

Heat exchanger materials must have good corrosion resistance in lead alloy coolant, particularly given the thin sections typically employed for such applications. Corrosion test requirements are similar to those for other core components, but without the requirement for radiation resistance.

For process heat applications associated with high temperature LFRs, an intermediate heat transport loop is probably needed to isolate the reactor from the energy converter for both safety assurance and product purity. Heat exchanger materials screening will be needed very early in the program for potential intermediate loop fluids, including molten salts, He, CO_2 and steam. For interfacing with thermochemical water cracking, the chemical plant fluid is HBr plus steam at 750ºC and low pressure. For interfacing with turbomachinery, the working fluid options are supercritical CO_2 or superheated or supercritical steam.

Corrosion resistance for candidate heat exchanger materials must be established. This may include corrosion resistance to lead alloys, high temperature supercritical carbon dioxide, and aqueous HBr solutions, and molten salt. Decisions on establishing this aspect of the LFR materials program will require better definition of system requirements.

VIII.3.4.6 Materials for LFR Balance-of-Plant Materials

For lower temperature LFRs, the energy production side is likely to be either a Rankine cycle or a Brayton cycle using supercritical carbon dioxide as the working fluid. No development is needed for the Rankine cycle, as this is commonly used in commercial energy production. Qualified materials for a supercritical Brayton cycles do not exist. If the proposed Ca-Br cycle is selected for for hydrogen production, materials qualified for HBr acid use will be chosen.

A key unknown is corrosion resistance in supercritical carbon dioxide for a Brayton cycle. Another is fabricating joints between heat exchangers and bromic acid containing piping.

VIII.3.4.7 Expected Research, Testing, and Qualification Needs for LFR Materials

Because qualification of any cladding, core internals, and vessel material requires significant corrosion testing, dedicated corrosion test loops capable of testing in lead and lead bismuth up to temperatures of 800°C are needed. Weight loss under typical temperature, coolant chemistry, and coolant velocity conditions must be ascertained, as must general corrosion. Weight loss as a function of exposure time in lead alloy is required for all candidates. Stress corrosion cracking and liquid metal embrittlement resistance must be demonstrated.

Radiation performance must be confirmed or established for any candidate cladding material. The database must establish the swelling, tensile, creep, toughness, and fatigue performance under the range of temperatures and doses expected by cladding

If development of supercritical carbon dioxide Brayton cycles or Ca-Br thermochemical cycles or the use of molten salts for heat transfer for hydrogen production are pursued, then corrosion resistance of all candidate materials must be evaluated in the appropriate media and those selected as primary candidates qualified.

VIII.4 MATERIALS FOR ENERGY-CONVERSION SYSTEM

The various approaches for energy conversion currently being considered within the Gen IV reactors include both electrical generation and use of process heat for hydrogen production. While many of the materials issues for electrical generation are similar to those in the fossil fuel industry, the same cannot be said for hydrogen production. Multiple approaches for nuclear hydrogen production include the use of thermo-chemical separation and thermally assisted electrolysis as the two leading candidate processes. Both of these approaches will have significant materials challenges including high-temperature structural stability, stability and effectiveness of special functional materials for catalysis and separation technology, thermal barrier materials, and materials compatibility with a variety of heat-transfer media and process-related chemicals.

Of particular concern are the very high-temperature heat exchangers envisioned both on the reactor side and the hydrogen production side of the process-heat transfer loop, as well as the lower temperature heat exchangers used within any chemical separation system. The combination of high-temperature operations and simultaneous exposure to multiple process and heat transfer fluids will present significant challenges to maintain the integrity of the thin sections inherent in heat exchangers.

While some of the requirements for the high-temperature materials will be addressed as part of the cross-cutting task described in section VIII.2 or within the R&D identified within the individual reactor systems, the remaining specialized materials requirements for energy conversion systems will need to be addressed separately. Those tasks that address the generation of electricity will continue to be conducted within the Gen IV Program itself. Those tasks that will address the production of hydrogen will fall under the newly established NHI Program

Therefore, activities within this subtask will focus on working with the energy conversion crosscutting leadership with the Gen IV and NHI programs to develop and implement an integrated materials program that addresses their needs for high-temperature materials, materials compatibility, corrosion, and functional materials in a coordinated, prioritized manner.

VIII.5 NATIONAL MATERIALS PROGRAM INTEGRATION

To help ensure that the materials R&D activities conducted within the overall Gen IV Reactor Initiative form an integrated, efficient program, an additional task is included to coordinate, prioritize, and manage materials cross-cutting research with that needed for each specific reactor concept and the energy-conversion system. Principal activities within this task will be to work with the product teams to:

- Develop a detailed understanding of the conditions that all major components and subsystems in each reactor concept and energy-conversion system must withstand (e.g. temperature, irradiation dose, corrosive media, etc., and their combinations);
- Collect and evaluate existing related data from domestic and foreign sources to determine deficiencies in materials data or capabilities;
- Provide cross-platform guidance to ensure appropriate materials R&D is performed in support of each reactor concept, with minimum overlap and no technical voids;
- Ensure that the cross-cutting materials research provides needed and useful information that can be applied to support all reactor concepts; and
- Help ensure that an integrated materials research is developed, prioritized, and implemented to address the materials needs of the overall Gen IV Reactor Initiative.

The major products of this task will be to provide initial and regularly updated reports assessing potential materials for use in all Gen IV reactor concepts and providing recommendations for reactor-specific materials screening and evaluations to identify viable candidate materials.

VIII.6 FY 2004 WORK SCOPE

Fiscal year 2004 scope for Gen IV materials R&D is divided into seven major areas. The tasks active in FY04 in this area and their funding levels are listed in Table 1, with a summary provided below by major area.

VIII.6.1 Qualification of Materials for Radiation Service

Initiate irradiation planning for commercial and near-commercial materials with emphasis on RPV materials and reactor internals for NGNP service. This will include the selection of a designated material test reactor for a high-temperature, low-flux vessel irradiation module and initiation of the facility's design. Detailed plans for Gen IV irradiation experiments will be prepared.

VIII.6.2 Qualification of Materials for High-Temperature Service

Establishment of a comprehensive, searchable database of existing information on materials properties relevant to Gen IV needs will be initiated. Sources will include the ASME Pressure Vessel and Piping Code, the Nuclear System Materials Handbook (NSMH), the fusion program, the fossil energy program, the space reactors program, and the accelerator materials program. Detailed plans for high-temperature Gen IV materials experiments will be prepared.

VIII.6.3 Microstructural Model Development

An analysis of the current state of the art in microstructural modeling will be conducted to assess the modeling needs of the Gen-IV reactor program and an initial report prepared.

VIII.6.4 High-Temperature Design Methodology

Prepare a detailed assessment of recent applicable high-temperature structural design methodology developments in the United States, Europe, and Japan. Prepare detailed reports on the assessment and on program plans. Conduct exploratory materials tests to establish key inelastic behavioral features of modified 9Cr-1Mo steel (Grade 92) and Alloy 617 at the high-end of the useful temperature range and initiate simplified analysis methods for meeting strain limits based on observed behavior and provide a report. Renew participation in the ASME Code Subgroup on Elevated Temperature Design.

VIII.6.5 Reactor-Specific Materials

Provide coordination and integration of specific materials needs of each reactor type to develop and implement materials R&D required to materials-related issues for each reactor concept. Specific tasks will include:

- Prepare NGNP materials program QA plans
- Complete irradiation of preliminary HFIR rabbits capsules and archive samples of NGNP candidate graphites
- Issue initial NGNP Materials Qualification and Selection Program Plan
- Issue revised NGNP Materials Qualification and Selection Program Plan
- Issue report of MRC review of NGNP Materials Program Plan
- Issue report on materials R&D plan for GFR
- Evaluate candidate SCWR internals alloys in Univ. of Wisc. supercritical water corrosion loop
- Construct a Constant Extension Rate Test (CERT) facility for the Univ. of Mich. hot cells for testing neutron irradiated candidate SCWR internals alloys
- Evaluate water chemistry control strategies in LWRs and fossil plants for applicability to the SCWR
- Perform scoping studies of lead-alloy coolant technology and materials compatibility using the DELTA loop for LFR
- Complete analysis of test specimens from DELTA 1000-hr corrosion test for LFR
- Prepare input for a multiyear LFR plan for coolant technology and materials compatibility
- Review corrosion control strategies for Pb and Pb-Bi coolants to select viable avenue or avenues for corrosion control for LFR
- Prepare revised LFR materials R&D plan

Only limited funding for task coordination and integration is allocated from within the materials crosscut for this activity. Funding for technical contributions is provided by the affected individual reactor systems.

VIII.6.6 Materials for Power Conversion Systems

Provide coordination and integration of specific materials needs of power conversion systems to develop and implement materials R&D required to address the materials-related issues for power conversion systems.

Only limited funding for task coordination and integration is allocated from within the materials crosscut for this activity. Funding for technical contributions will be provided by the affected individual reactor systems and the power conversion NTD.

VIII.6.7 National Materials Program Coordination & Management

Provide overall management, coordination, and prioritization of the integrated Gen IV National Materials Program and prepare updated, interim report on assessment and selection of crosscutting candidate materials for high-temperature and radiation service in Gen IV reactor systems.

Table 1 Summary of Funding for Crosscutting Materials Tasks for FY 2004 (K$).

VIII.7 FY 2005 WORK SCOPE

Fiscal year 2005 scope for the materials crosscut is divided into seven major areas. The tasks active in FY05 in this area and their funding levels (both required and actual) are listed in Table 2. Because the fiscal year 2005 funding level may not be sufficiently large to enable all the required scope to be completed, the following subsections describe first the required scope for each task followed by a description of the scope that can be accomplished with the funding currently projected to be available for that same task.

VIII.7.1 Qualification of Materials for Radiation Service

Required Scope: Initiate scoping irradiations of commercial and near-commercial materials on RPV materials and reactor internals for VHTR service and initiate similar irradiations on materials for VHTR control rods and insulation, as well as advanced alloys and materials for other concepts. Continue the design of the high-temperature vessel irradiation module at the material test reactor designated in FY2004.

Actual Scope: Continue the design of the high-temperature vessel irradiation module at the material test reactor designated in FY2004.

VIII.7.2 Qualification of Materials for High-Temperature Service

Required Scope: Complete the initial establishment of the database of existing information on materials properties relevant to Gen IV. Initiate high-temperature screening studies on commercial and near-commercial alloys. Identify deficiencies in high-temperature materials needed for ASME codification.

Actual Scope: Continue initial establishment of the materials database.

VIII.7.3 Microstructural Model Development

Required Scope: Initiate the development of detailed microstructural models for radiation service and high-temperature exposure of structural materials in critical areas focusing on mechanisms responsible for the development of radiation-enhanced, -induced, and -modified microstructural changes and modeling of the nucleation-phase of the generalized class of defects that are produced during irradiation. Prepare integrated report that prioritizes cross-cutting microstructural modeling needs for Gen-IV reactor program, and identifies needed special-purpose experiments.

Actual Scope: Prepare integrated report that prioritizes cross-cutting microstructural modeling needs for Gen-IV reactor program and identifies needed special-purpose experiments.

VIII.7.4 High-Temperature Design Methodology

Required Scope: Initiate development of rules to allow use of low-temperature criteria for limited high-temperature operation (excursions) of the representative pressure vessel material. Initiate uniaxial and biaxial exploratory deformation testing and constitutive equation development for modified 9Cr-1Mo steel (Grade 92) and Alloy 617 and provide interim constitutive equations for them. Initiate structural deformation and failure tests of Alloy 617 models at very high temperatures.

Actual Scope: Initiate development of rules to allow use of low-temperature criteria for limited high-temperature operation (excursions) of the representative pressure vessel material.

VIII.7.5 Reactor-Specific Materials

Required Scope: Provide coordination and integration of specific materials needs of each reactor type to develop and implement materials R&D required to materials-related issues for each reactor concept. Specific tasks will include:

- Prepare detailed NGNP material candidate selection documents
- Evaluate the potential effects of low damage rate neutron environments on the long term, high temperature microstructural stability of candidate alloys for NGNP RPV and metallic core components
- Based on an analysis of spectral and flux distributions for the NGNP metallic core components, define the nature and magnitude of potential radiation effects on the performance of candidate alloys
- Continue design and fabrication of irradiation facilities and fabrication of test specimens for NGNP
- Initiate preliminary irradiations of potential RPV candidate alloys in high flux experiments for NGNP
- Initiate procurement of structural alloys needed for the validation or extension of construction codes to the NGNP design conditions.
- Initiate time-independent mechanical properties evaluation of commercial and near-commercial alloys for NGNP service.
- Complete post irradiation examination of NGNP candidate graphite from HFIR rabbit capsules
- Purchase initial quantities of NGNP candidate graphites
- Initiate design and construction of NGNP materials compatibility test facilities and establish required test matrices.
- Initiate emissivity testing for NGNP
- Initiate materials compatibility studies of ODS ferrritic-martensitic steels, Nb- and Mo-base alloys and ceramics including Nb- and Mo-base cermets with impure helium for GFR
- Initiate compilation of available information on solubility of SCWR power conversion systems candidate materials in supercritical steam

- Continue evaluation of candidate SCWR internals alloys in Univ. of Wisc. supercritical water corrosion loop Univ. of Mich. CERT
- Establish the extent of additional materials testing needs for compatibility with Pb and Pb-Bi for LFR
- Complete scoping studies of preliminary LFR candidate materials for corrosion resistance
- Complete scoping studies of surface treatments for controlling corrosion in LFR environments

Actual Scope: Provide coordination and integration of specific materials needs of each reactor type to develop and implement materials R&D required to address the graphite- and materials-compatibility-related issues for each reactor concept. Specific tasks to be performed will be developed in cooperation with the reactor SIMs once their actual materials funding for FY05 has been developed.

Only limited funding for task coordination and integration is allocated from within the materials crosscut for this activity. Funding for technical contributions is provided by the affected individual reactor systems.

VIII.7.6 Materials for Power Conversion Systems

Required Scope: Provide coordination and integration of specific materials needs of power conversion systems to develop and implement materials R&D required to address the materials-related issues for power conversion systems. Detailed tasks are not yet defined.

Actual Scope: Provide coordination and integration of specific materials needs of power conversion systems to develop and implement materials R&D required to address the materials-related issues for power conversion systems. Detailed tasks are not yet defined.

Only limited funding for task coordination and integration is allocated from within the materials crosscut for this activity. Funding for technical contributions will be provided by the affected individual reactor systems and the power conversion NTD.

VIII.7.7 National Materials Program Coordination & Prioritization

Required Scope: Provide overall management, coordination, and prioritization of the integrated Gen IV National Materials Program and prepare updated, interim status on assessment and selection of crosscutting candidate materials for high-temperature and radiation service in Gen IV reactor systems.

Actual Scope: Provide overall management, coordination, and prioritization of the Gen IV National Materials Program and prepare updated, status report on assessment and selection of crosscutting candidate materials for high-temperature and radiation service in Gen IV reactor systems.

Table 2 Summary of Funding for Crosscutting Materials Tasks for FY 2005 ($K).

VIII.8 TEN-YEAR PROGRAM PLAN

The high-level ten-year objectives of the National Materials R&D program within the Gen IV Initiative are to:

- Complete an assessment of cross-cutting and reactor-specific materials for use in all Gen IV reactor concepts to identify viable candidate materials;
- Complete the initial development of a comprehensive irradiation-effects materials database for materials needed for radiation service in Gen IV reactors;
- Complete initial development of a comprehensive high-temperature materials properties database to support the design, use, and codification of materials needed for Gen IV reactors;
- Complete adequate qualification of the materials to be used in the NGNP reactor to enable the design and ordering of all major components and subsystems;
- Complete initial development of an improved high-temperature design methodology that will support design, use, and codification of materials needed for Gen IV reactors;
- Complete development of an interim comprehensive model for predicting long-term properties of materials needed for Gen IV reactors as a function of thermal and irradiation exposure; and
- Interface with Gen IV International Forum and relevant domestic and foreign materials research programs to optimize the effectiveness of materials R&D plan

The anticipated deployment of the NGNP in 2017 will require a strong acceleration of materials qualification needed to enable design and ordering of long-lead components by about 2009. As a result, a major focus of materials research during the next ten years will be on the qualification of commercial and near-commercial materials and the related high-temperature design methodology needed to specify and order those components. Parallel studies on materials for other reactor concepts will both take advantage of the accelerate work for the NGNP and examine additional materials under other conditions where the NGNP materials studies are inadequate or inappropriate for their conditions. To help level required resources to the extent possible, the additional studies on materials for other reactor concepts will generally increase in scope as portions of the NGNP-related materials studies are completed.

The milestones of the ten-year plan are as follows:

FY 2004

- Survey and select a material test reactor for low-flux pressure vessel irradiations
- Prepare detailed plans for irradiation experiments
- Initiate irradiation specimens fabrication and facility design
- Initiate compilation of database on Gen IV materials
- Prepare detailed plans for high-temperature materials experiments

- Publish initial report on microstructural modeling needs and prepare detailed interim report assessing current status of microstructural model development and supporting microstructural analysis.
- Prepare detailed assessment report of latest international developments in high-temperature structural design.
- Prepare detailed high-temperature design methodology program plan.
- Conduct exploratory deformation tests on modified 9Cr-1Mo steel (grade 92) and Alloy 617 and initiate development of simplified methods for satisfying Code strain limits.
- Reestablish participation in the ASME Subgroup on Elevated Temperature Design.
- Prepare NGNP materials program QA plans
- Complete irradiation of preliminary HFIR rabbits capsules and archive samples of NGNP candidate graphites
- Issue initial NGNP Materials Qualification and Selection Program Plan
- Issue revised NGNP Materials Qualification and Selection Program Plan
- Issue report of MRC review of NGNP Materials Program Plan
- Issue report on materials R&D plan for GFR
- Evaluate candidate SCWR internals alloys in Univ. of Wisc. supercritical water corrosion loop
- Construct a Constant Extension Rate Test (CERT) facility for the Univ. of Mich. hot cells for testing neutron irradiated candidate SCWR internals alloys
- Evaluate water chemistry control strategies in LWRs and fossil plants for applicability to the SCWR
- Perform scoping studies of lead-alloy coolant technology and materials compatibility using the DELTA loop for LFR
- Complete analysis of test specimens from DELTA 1000-hr corrosion test for LFR
- Prepare input for a multiyear LFR plan for coolant technology and materials compatibility
- Review corrosion control strategies for Pb and Pb-Bi coolants to select viable avenue or avenues for corrosion control for LFR
- Prepare revised LFR materials R&D plan
- Prepare updated, interim report on assessment and selection of crosscutting candidate materials for high-temperature and radiation service in Gen IV reactor systems.

FY 2005

- Continue design of facilities for both low flux and high flux high temperature irradiations
- Initiate low-dose scoping irradiations of commercial and near-commercial materials with emphasis on materials needs for VHTR service
- Complete establishment of initial database for candidate materials for high-temperature and radiation service for all Gen IV reactor systems
- Identify deficiencies in high-temperature materials needed for codification

- Initiate mechanical testing of high-temperature materials
- Prepare integrated report that prioritizes cross-cutting microstructural modeling needs for Gen-IV reactor program, and identifies needed special-purpose experiments
- Initiate microstructural model development in critical areas
- Initiate development of rules to allow use of low-temperature design criteria for vessels subjected to limited high-temperature service.
- Provide interim constitutive equations for modified 9Cr-1Mo steel (Grade 92) and Alloy 617.
- Initiate uniaxial and biaxial creep-fatigue tests and development of creep-fatigue damage model for modified 9Cr-1Mo steel (Grade 92) and Alloy 617.
- Initiate structural tests of Alloy 617 models at very high temperatures.
- Prepare detailed NGNP material candidate selection documents
- Evaluate the potential effects of low damage rate neutron environments on the long term, high temperature microstructural stability of candidate alloys for NGNP RPV and metallic core components
- Based on an analysis of spectral and flux distributions for the NGNP metallic core components, define the nature and magnitude of potential radiation effects on the performance of candidate alloys
- Continue design and fabrication of irradiation facilities and fabrication of test specimens for NGNP
- Initiate preliminary irradiations of potential RPV candidate alloys in high flux experiments for NGNP
- Initiate procurement of structural alloys needed for the validation or extension of construction codes to the NGNP design conditions.
- Initiate time-independent mechanical properties evaluation of commercial and near-commercial alloys for NGNP service.
- Complete post irradiation examination of NGNP candidate graphite from HFIR rabbit capsules
- Purchase initial quantities of NGNP candidate graphites
- Initiate design and construction of NGNP materials compatibility test facilities and establish required test matrices.
- Initiate emissivity testing for NGNP
- Initiate materials compatibility studies of ODS ferrritic-martensitic steels, Nb- and Mo-base alloys and ceramics including Nb- and Mo-base cermets with impure helium for GFR
- Initiate compilation of available information on solubility of SCWR power conversion systems candidate materials in supercritical steam
- Continue evaluation of candidate SCWR internals alloys in Univ. of Wisc. supercritical water corrosion loop Univ. of Mich. CERT
- Establish the extent of additional materials testing needs for compatibility with Pb and Pb-Bi for LFR
- Complete scoping studies of preliminary LFR candidate materials for corrosion resistance

- Complete scoping studies of surface treatments for controlling corrosion in LFR environments
- Prepare updated, status report on assessment and selection of crosscutting candidate materials for high-temperature and radiation service in Gen IV reactor systems.

FY 2006

- Initiate low-dose scoping irradiations of advanced materials
- Complete initial low-dose scoping irradiations of commercial and near-commercial materials
- Complete design of facilities for both low flux and high flux irradiations
- Initiate joining and combined-effects screening studies on commercial and near-commercial alloys.
- Prepare documents of 316FR and alloy 617 for ASME codification.
- Prepare interim report on results of model-based the nucleation phase of the significant extended defects produced under irradiation.
- Use interior constitutive equations to develop isochronous stress-strain curves and other predicted behavioral representations for modified 9Cr-1Mo steel (Grade 92) and Alloy 617.
- Complete design and fabrication of primary irradiation facility for low flux irradiations for NGNP
- Complete preliminary irradiations and PIE of potential candidate alloys in high flux experiments for NGNP
- Initiate irradiations of preliminary candidate RPV alloys in the low flux irradiation facility and initiate irradiation experiments of metallic internals alloys with a high thermal to fast flux for NGNP
- Select primary high-temperature materials and complete planning needed to qualify alternate materials for NGNP structural components.
- Initiate mechanical testing of CCM, insulator, metallic reactor internals, bolting, and IHX materials in the NGNP gaseous environment.
- Complete initial assessment and provide materials use guidelines for NGNP HX materials.
- Transition constitutive equation development to candidate NGNP pressure boundary materials and NGNP very-high-temperature component materials.
- Develop initial simplified high-temperature design rules for use in preliminary design of NGNP components.
- Complete ASTM standard materials specification development in support of NGNP graphite
- Complete purchase of pre-production lot(s) of NGNP candidate graphite(s)
- Complete preliminary characterization of baseline physical and mechanical properties of NGNP graphite
- Complete design and construction of NGNP graphite irradiation creep HFIR capsules
- Evaluate capabilities of suppliers for thick-section RPV for SCWR
- Initiate demonstration of fabrication capabilities needed for SCWR

- Begin compilation, evaluation and new testing for unirradiated mechanical properties data for reactor internals for SCWR
- Complete compilation of available information on solubility of SCWR power conversion systems candidate materials in supercritical steam
- Initiate measurements of solubility of SCWR candidate materials in supercritical steam
- Initiate evaluation of factors affecting steam condensation and stability of corrosive species in SCWR power conversion systemss
- Continue corrosion and SCC testing of primary SCWR candidate materials for core support components in supercritical water
- Initiate corrosion fatigue testing for SCWR pump materials in supercritical water
- Develop detailed integrated LFR materials corrosion testing and evaluation plan
- Complete shakedown and begin full operation of Pb and Pb-Bi test facilities
- Complete initial assessment of creep and aging mechanisms in LFR materials
- Complete initial assessment of surface protection mechanisms in LFR materials
- Complete selection of primary candidate materials for LFR system
- Prepare updated, status report on assessment and selection of crosscutting candidate materials for high-temperature and radiation service in Gen IV reactor systems

FY 2007

- Complete low-dose scoping irradiations of advanced materials, complete low dose irradiations, and complete PIE of commercial and near-commercial materials
- Complete selection of primary RPV candidate materials based on screening irradiation experiments.
- Continue studies of time-dependent mechanical properties combined-effects on commercial and near-commercial alloys.
- Continue detailed studies of high-temperature, time-dependent properties for advanced candidate materials for high-temperature service and required materials modifications
- Continue joining studies on commercial and near-commercial alloys and initiate joining studies on advanced high-temperature materials
- Prepare interim report describing overall microstructural evolution under low and high temperature irradiation, include results from preliminary modeling studies and microstructural characterization.
- Prepare interim report on mechanisms responsible for the development of radiation-enhanced, -induced, and -modified microstructural changes.
- Propose creep-fatigue criteria for modified 9Cr-1Mo steel (Grade 92) and Alloy 617.
- Complete Alloy 617 confirmatory structural tests, and initiate testing of models for other key Gen IV structural materials.
- Initiate irradiation experiments of candidate materials for internal VHTR structures.
- Continue low flux irradiations of preliminary candidate RPV alloys for NGNP

- Continue irradiation experiments of candidate metallic internals alloys with a high thermal to fast flux for NGNP
- Prepare report on results of irradiations of potential candidate alloys in high flux experiments for NGNP
- Evaluate need for, and role of, exemption rules for high-temperature design of proposed NGNP pressure vessel and very-high-temperature component materials; develop needed rules.
- Complete graphite physical and mechanical properties evaluations for NGNP
- Complete preliminary graphite oxidation effects studies of NGNP graphites
- Complete preliminary irradiation effects studies of NGNP graphites
- Complete evaluation of C-C composites for control rods, bolting, and insulation materials for NGNP
- Complete C-C composite as-received properties evaluation for NGNP
- Initiate mechanical testing of pressure boundary and insulation materials in the NGNP gaseous environment.
- Complete preliminary evaluations of materials compatibility for NGNP applications
- Initiate materials compatibility studies with super-critical CO_2 in the temperature range of 400 to 650°C for the GFR
- Begin fabrication of irradiation experiments for reactor internals prime candidate materials for SCWR
- Prepare progress report on demonstration of RPV fabrication capabilities for SCWR
- Prepare final report on unirradiated mechanical properties for SCWR
- Complete corrosion and SCC screening tests in supercritical water for SCWR
- Complete measurements of solubility of SCWR candidate materials in supercritical steam
- Initiate corrosion and SCC testing SCWR power conversion systems materials in supercritical water
- Complete evaluation of factors affecting condensation and stability of corrosive species in SCWR power conversion systemss
- Complete assessment of mechanical and corrosion properties of primary candidate LFR materials in as-received condition
- Initiate aging and irradiation assessment of primary candidate LFR materials
- Prepare report on overall assessment and interim selection of assessment and selection of crosscutting candidate materials for high-temperature and radiation service in Gen IV reactor systems.

FY 2008

- Initiate high-dose scoping irradiations of advanced materials for reactor internals
- Prepare report on scoping studies of low-dose irradiations of commercial, near-commercial, and advanced materials for radiation service

- Complete preliminary assessment of candidate materials for radiation service for high temperature reactors and provide input to remaining reactor concepts regarding establishing detailed plans to meet their needs
- Prepare interim report on initial studies of time-dependent mechanical properties, combined-effects, and joining technology for advanced alloys and provide recommendations for further studies.
- Prepare interim report on kinetics and thermodynamics of formation and stability of the very fine oxide clusters in ODS alloys, and make recommendations on use of ODS alloys.
- Prepare interim report on microstructural basis for mechanisms that contribute to high-temperature, time-dependent damage.
- Recommend interim unified constitutive equations for down-selected Gen IV high-temperature materials.
- Initiate development of a very-high-temperature flaw assessment procedure.
- Continue irradiations of preliminary candidate RPV alloys for NGNP
- Continue irradiations of candidate metallic internals alloys for NGNP
- Complete assessment of alternate fabrication methods for the NGNP RPV and recommend materials and fabrication processes.
- Complete plan and initiate research to ensure the integrity of NGNP components for a design life beyond 300,000 hours.
- Complete final characterization of baseline physical and mechanical properties of NGNP graphites
- Complete final characterization of baseline physical and mechanical properties of NGNP graphites
- Complete SiC-SiC composite as-received properties evaluation for NGNP
- Complete C-C composite irradiated properties evaluation for NGNP
- Complete ceramic insulation materials as-received properties evaluation for NGNP
- Initiate mechanical testing of turbine and recuperator materials in the NGNP gaseous environment.
- Complete final evaluations of materials compatibility for NGNP applications except for C-C composites
- Complete initiate evaluation of ODS ferrritic-martensitic steels, Nb- and Mo-base alloys and ceramics including Nb- and Mo-base cermets compatibility studies with impure helium for GFR.
- Assess higher strength RPV steels together with manufacturers capabilities, prepare report for SCWR
- Begin fabrication studies of higher strength RPV steels for SCWR
- Initiate corrosion and SCC screening tests with simulated in-reactor chemistry for SCWR
- Complete fabrication and place first irradiation experiments into reactor for SCWR
- Complete fabrication and place first irradiation experiments into in-reactor supercritical water loop for SCWR

- Complete corrosion and SCC testing of primary candidate materials for SCWR core support components in supercritical water
- Initiate post-irradiation corrosion and IASCC testing of SCWR core support materials in supercritical water
- Initiate corrosion fatigue testing for valve materials in supercritical water at simulated chemistry for SCWR
- Complete collection and evaluation of solid particle erosion in supercritical steam from fossil experience for SCWR applicability
- Initiate testing to predict oxide scale growth, frequency and mode of scale spallation of SCWR power conversion systems materials
- Complete initial phase of aging and irradiation resistance assessment of primary candidate LFR materials
- Prepare updated, status report on qualification of crosscutting candidate materials for high-temperature and radiation service in Gen IV reactor systems

FY 2009

- Complete low-dose irradiation experiments and PIE of advanced materials for reactor internals
- Complete irradiation experiments of control structural elements for high temperature reactors.
- Initiate qualification testing of advanced materials for high-temperature service for all advanced reactor concepts
- Complete initial qualification studies of advanced materials for high-temperature service and provide recommendations for further studies for all advanced reactor concepts
- Prepare interim report on atomistic modeling in support of advanced micromechanical models for predicting mechanical properties of structural materials.
- Prepare report on results of comprehensive modeling of radiation-induced microstructural evolution in the primary Gen-IV candidate structural materials, identify areas for further model development.
- Recommend final revised simplified methods for satisfying strain limits and creep-fatigue criteria in high-temperature structural design.
- Complete PIE of preliminary candidate RPV alloys and prepare report on results for NGNP
- Initiate irradiations of primary candidate RPV alloys for NGNP
- Complete PIE of candidate metallic internals alloys for NGNP
- Deliver materials performance data suitable for interim use in the detailed design of the NGNP RPV, SCS, and hot ducting.
- Complete graphite irradiation effects studies
- Complete graphite design model modification and verification
- Complete graphite codes and standards development
- Complete final graphite oxidation effects studies of NGNP graphites
- Complete graphite irradiation effects of properties studies of NGNP graphites

- Complete SiC-SiC composite irradiated properties evaluation for NGNP
- Complete ceramic insulation irradiated properties evaluation for NGNP
- Complete mechanical testing of CCM, insulator, metallic reactor internals, turbine and recuperator materials in the NGNP gaseous environment for a design life of up to 300,000 hours
- Complete preliminary evaluation of emissivity studies for NGNP
- Complete final evaluations of materials compatibility for C-C composites for NGNP
- Complete development of materials design data needed to order major NGNP components.
- Complete initiate evaluation of materials compatibility studies with super-critical CO_2 for the GFR.
- Initiate radiation effects and thermal aging studies of higher strength RPV steels for SCWR
- Complete testing and write final report on corrosion and SCC tests on unirradiated materials in supercritical water for SCWR
- Remove first low-dose core structural material specimens from reactor, evaluate mechanical properties and corrosion for SCWR
- Initiate irradiation of SCWR candidate materials in supercritical pumped flow loop, post-irradiation mechanical properties testing, microstructural characterization, and corrosion and IASCC testing in supercritical water
- Complete design database for short-term mechanical and corrosion properties of primary candidate LFR materials in as-received condition
- Prepare updated, status report on qualification of crosscutting candidate materials for high-temperature and radiation service in Gen IV reactor systems.

FY 2010

- Complete initial irradiation experiments of advanced materials for reactor internals
- Complete PIE of control structural elements for high temperature reactors
- Continue qualification testing of advanced materials (such as ODS etc.) for high-temperature service for all advanced reactor concepts
- Prepare detailed interim report on integrated models for assessing radiation-induced and time-dependent, high-temperature changes in Gen-IV candidate structural materials.
- Provide improved models of mechanisms for high-temperature, time-dependent plasticity as input to the formulation of design criteria for elevated temperature VHTR materials service.
- Finalize constitutive equations for all key Gen IV structural metals.
- Complete irradiation experiments of RPV and insulation materials and complete report on recommendations for application of selected materials for VHTR radiation service
- Prepare report on results of candidate metallic internals alloys irradiation experiments for NGNP

- Validate creep-fatigue criteria for down-selected NGNP component materials and weldments.
- Complete confirmatory structural tests and assessments for key NGNP structural alloys.
- Complete irradiation creep studies of NGNP graphites
- Complete materials compatibility studies for NGNP
- Complete emissivity studies for NGNP
- Complete structural composite aged properties evaluation for NGNP
- Complete ceramic insulation aged properties evaluation for NGNP
- Prepare progress report on fabrication results for thick section fabrication on conventional RPV steels for SCWR
- Place specimens of higher strength RPV steels in reactor for SCWR
- Complete final report on corrosion and SCC tests with simulated in-reactor chemistry for SCWR
- Remove intermediate dose structural material specimens from reactor, carry out extensive mechanical property, microstructural and corrosion characterizations for SCWR
- Complete design database for effects of aging and irradiation on primary candidate LFR materials
- Complete materials compatibility viability studies for LFR

FY 2011

- Complete high dose scoping irradiations of advanced materials for reactor internals
- Complete assessment of candidate advanced materials for high dose internals radiation service
- Complete assessment and provide report on irradiations of control structural elements
- Continue qualification testing of advanced materials (such as alloy 214) for high-temperature service for all advanced reactor concepts.
- Prepare final report on model-based analysis of formation and stability of radiation-induced or enhanced phase stability in irradiated alloys, including oxide clusters in ODS alloys.
- Prepare final report on results of microstructural analysis of irradiated and thermally-aged Gen-IV candidate structural materials examined under this task.
- Validate final simplified design rules for ratcheting and creep-fatigue damage for Gen IV materials.
- Initiate development of procedures to guard against thermal-striping failures in upper internal structures of LFR.
- Continue mechanical properties assessment of materials in specific component forms and reconcile reference and component-forms-specific databases for NGNP.
- Complete irradiations and PIE of RPV alloys for NGNP
- Prepare final report on results of RPV alloys irradiation experiments for NGNP

- Complete demonstration of fabrication capability for thick-section conventional RPV steels, write comprehensive report for SCWR
- Write progress report on testing of higher strength RPV steels for SCWR
- Prepare progress report on mechanical property, microstructural and corrosion characterizations of intermediate dose structural material specimens for SCWR
- Compare mechanical property, microstructural and corrosion results from materials irradiated conventionally and those irradiated in SC water for SCWR

FY 2012

- Initiate primary high dose irradiations of candidate advanced materials for internals radiation service
- Prepare reports on scoping irradiations of advanced materials for reactor internals
- Prepare final report on micromechanical models, including their atomistic basis, used to predict relationship between microstructure and mechanical properties in structural materials planned for use in Gen-IV reactor program.
- Prepare high-temperature materials supporting documents for licensing.
- Assess mechanical properties of reactor-specific materials.
- Continue assessment mechanical properties of special and advanced materials.
- Complete development of flaw assessment procedure for Gen IV structural alloys.
- Develop and prepare report on recommendations for RPV surveillance program for NGNP
- Prepare final report on all tests of higher strength of RPV steels for SCWR
- Initiate development of procedures for cladding of higher strength steels for SCWR
- Remove high dose prime candidate materials from reactor, carry out extensive mechanical property, microstructural and corrosion characterizations for SCWR
- Compare results from high and intermediate dose core structural materials and prepare interim report for SCWR

FY 2013

- Continue high dose irradiations of candidate advanced materials for internals radiation service and provide recommendations for further studies for all advanced reactor concepts
- Assess mechanical properties of cladding materials.
- Continue to assess mechanical properties of reactor-specific materials.
- Investigate time-dependent crack growth properties of crosscutting materials.
- Prepare final report on integrated models for assessing radiation-induced and time-dependent, high-temperature changes in Gen-IV candidate structural materials and provide recommendations for any further studies required to refine and validate the models in support of Gen-IV reactor operations.
- Recommend thermal-striping assessment guidelines.
- Resolve identified shortcomings, issues, and regulatory concerns in high-temperature structural design methodology.

- Complete final reports on radiation effects and thermal aging of higher strength RPV specimens and on procedures for cladding of higher strength RPV steels for SCWR
- Complete final reports on mechanical property, microstructural and corrosion evaluations of irradiated prime candidate materials in conventional post-test and in-situ SC water environments for SCWR
- Complete testing to predict oxide scale growth, frequency and mode of scale spallation of SCWR power conversion systems materials
- Complete initial matrix of irradiation of SCWR candidate materials in supercritical pumped flow loop, post-irradiation mechanical properties testing, microstructural characterization, and corrosion and IASCC testing in supercritical water

VIII.9 PERFORMANCE MEASURES FY2005 – FY2013

The high-level performance measures for the Gen IV Reactor Materials Program are:

- Complete establishment of initial database for candidate materials for high-temperature and radiation service for all Gen IV reactor systems 9/05
- Complete initial assessment of candidate graphites for irradiation service in the NGNP reactor 9/07
- Complete preliminary assessment of candidate materials for high-temperature and radiation service for all Gen IV reactor systems and issue recommendations for final qualification 9/08
- Complete recommended revised simplified methods for satisfying strain limits and creep-fatigue criteria in high-temperature structural design. 9/09
- Complete development of materials design data needed to order major NGNP components. 9/09
- Prepare final report on micromechanical models used to predict relationship between microstructure and mechanical properties in structural materials for use in Gen-IV reactor program 9/12
- Resolve identified shortcomings, issues, and regulatory concerns in high-temperature structural design methodology 9/13

VIII.10 FUNDING REQUIREMENTS

Major deliverables for all crosscutting activities are supported by the funding shown in Table 3. Funding for reactor-specific and energy-conversion-system materials needs are included in the other reactor sections and the energy-conversion section.

Table 3 Summary of Required Funding for Crosscutting Materials Tasks for FY 2004 through FY 2013 (K$).